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COST/BENEFIT ANALYSIS OF GSI-15: RADIATION EFFECTS ON REACTOR VESSEL SUPPORTS

Final Report

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ABSTRACT

This report provides a cost/benefit (value/impact) analysis for Generic Safety Issue 15 (GSI-15). It assesses the core damage frequency and the risk associated with neutron embrittlement of the reactor pressure vessel supports (RPVSs). Five options for the resolution of GSI-15 are also evaluated. It then calculates the cost/benefit ratio that would result from implementation of any of the proposed options.

SUMMARY

Generic Safety Issue 15 (GSI-15) is concerned with neutron irradiation of the reactor pressure vessel supports (RPVSs). Neutron irradiation of structural materials causes embrittlement that may increase the probability of material failure due to a propagation of pre-existing flaws. The potential for neutron embrittlement of the RPVSs could be greater than was formerly anticipated. This report estimates the core damage frequency and the risk associated with RPVS failure, the cost involved in implementing any of five proposed resolutions, and the cost/benefit ratio that would be realized by implementation of each of the alternatives.

The five options proposed as resolutions for GSI-15 include: shielding the RPVSs from neutron irradiation, increasing the RPVS's operating temperature above the NDTT, replacing the RPVSs, heating the RPVSs sufficiently to anneal out any embrittlement, and strengthening or adding additional RPVSs.

The results indicate the estimated per plant costs range from a low value of $920,000 to increase the operating temperature of the supports to a high value of $89,000,000 to replace the existing supports. The low value takes into account averted onsite costs and assumes no replacement power would need to be purchased. The high value takes into account averted onsite costs, but assumes replacement power would have to be purchased for a 20-week period.

The results of the benefit analysis indicate a per-plant offsite dose risk of 2.9 person-rem/year of remaining reactor lifetime. This risk includes all the risk associated with support failure after embrittlement occurs. It was assumed that the implementation of any of the proposed options would remove 100% of the risk associated with failure of an embrittled support. The core damage frequency was found to be 8.8×10^{-5}/yr. This information provided cost/benefit ratios ranging from $5,300 per person-rem to $3,100,000 per person-rem.

CONTENTS

ACRONYMS

AOSC	averted onsite costs
BWR	boiling water reactor
COV	coefficient of variation
CSDSF	chemical shut down system failure
DPR	dollar to person-rem averted ratio
ECCSF	emergency core cooling system failure
EEDB	energy economic data base
EF	error factor
EFPY	effective full power years
GSI-15	Generic Safety Issue 15
LBLOCA	large break loss-of-coolant accident
LOCA	loss-of-coolant accident
LWR	light water reactor
NDTT	nil ductility transition temperature
NNB	no net benefit
NRC	Buclear Regulatory Commission
ORNL	Oak Ridge National Laboratory
PRA	probabilistic risk assessment
PWR	pressurized water reactor
QA/QC	quality assurance/quality control
RCF	reactor containment failure
RCS	reactor cooling system
RP	replacement power
RPSF	reactor protection system failure
RPV	reactor pressure vessel
RPVS	reactor pressure vessel support
RPVSF	reactor pressure vessel support failure
SBLOCA	small break loss-of-coolant accident
SF-PSD	safe plant shut down
SPRA	standard probabilistic risk assessment
SSE	safe shutdown earthquake

COST/BENEFIT ANALYSIS OF GSI-15: RADIATION EFFECTS ON REACTOR VESSEL SUPPORTS

1. INTRODUCTION

Neutron irradiation of structural materials causes embrittlement that may increase the probability of material failure due to a propagation of pre-existing flaws. In April 1988 data produced by Oak Ridge National Laboratory[1] (ORNL) suggested that the potential for neutron embrittlement of reactor pressure vessel supports (RPVS) could be greater than was formerly anticipated.

The first part of this report estimates the core damage frequency and risk associated with RPVS failure. The second part of this report presents the cost/benefit ratio for implementation of any of five solutions.

Normally the potential for brittle fracture in a material is quantified in terms of the material's nil ductility transition (NDT) temperature. The NDT temperature for a material is the temperature at which the material becomes prone to brittle failure. If the material is kept at a higher operating temperature than its NDT temperature, brittle fracture of the material will be prevented. The possible corrective measures to the damaged pressure vessel supports would fall in one of five categories:

- The supports can be shielded to reduce the neutron radiation exposure.
- The operating temperature of the supports can be increased above the NDTT of the support material.
- The embrittled supports can be replaced.
- The supports can be heated such that the embrittlement is annealed out.
- The embrittled supports can be left in place and additional supports can be added.

The RPVS designs for light water reactors (LWR) have been divided into five different categories (Reference 1). The support categories are skirt, long column, shield tank, short column, and suspension. The skirt type supports are located far enough away from the reactor core such that embrittlement induced failure of the support is not anticipated. All operating boiling water reactors (BWRs) except Big Rock Point have skirt type supports; therefore, they are not included in this study. Big Rock Point Nuclear Plant is the only operating plant with suspension type supports; it was not included in this study because of its small size (240 MWt) and low surrounding population density.

Table 1 lists the support type and the number of PWRs of each type in use. Since the skirt type supports are not likely to fail due to neutron embrittlement, they are removed from further consideration leaving 76 plants with susceptible supports.

Table 1. PWR reactor pressure vessel support utilization.

Support type	Number in use
Skirt	7
Long Column	11
Short Column	57
Shield Tank	8
Total susceptible plants	76

The analysis first estimated the core damage frequency and the risk associated with operating the 76 PWRs with possible radiation damaged RPVSs. It is assumed that any one of the 76 PWRs could have suspect RPVSs. Therefore, the event tree analysis was very conservative to be able to bound the different failure modes for the four different support types. Also, it was assumed that modifying the supports would reduce the embrittlement risk.

The second part of the analysis estimated the costs associated with fixing the different support types. The reduction in risk is understood to be the benefit, while the expenditure in fixing the supports is the cost. The cost/benefit ratio is then used as a basis for recommending what action should be taken. Consideration is also given to the core damage frequency resulting from embrittled supports.

2. BENEFIT EVALUATION

The benefit is defined as the reduction in risk obtained by fixing the neutron embrittled RPVS. To estimate the risk, two different scenarios were considered that could fail the supports. Event trees for each scenario were developed to obtain the associated probability of RPVS failure. The probability of RPVS failure was then multiplied by the associated consequence of the failure, thus obtaining the failure risk.

2.1. Event Tree Analysis

The GSI-15 event tree evaluation involves two different scenarios. The first scenario is a safe shutdown earthquake (SSE) as an initiating event and the potential failure of the RPVSs. The second scenario involves a small break loss-of-coolant-accident (SBLOCA) as the initiating event. The discussion of the scenarios includes the associated event tree and a detailed explanation of each event contained in the event tree.

Typical event tree methodology is used in the generation of the scenario event trees. At each branch node, the downward path represents the failure event that is listed above that node, while the upward path symbolizes the complement of the failure event. Each failure event portrays a phase in the scenario development and represents the failure of a particular safety function. Human errors and procedural guideline flaws are not incorporated into the event tree model.

The sequence outcomes are grouped into one of seven different categories. Table 2 lists the different categories along with a description of each category. The offsite release categories are taken from the WASH-1400[2] reactor safety report and classify various degrees of radioactive releases from containment. Each sequence was assigned to the offsite release category that best modeled its outcome. When a sequence could fall into one or more release categories (i.e., PWR 1 or PWR2, PWR 3 or PWR 4, etc.), the most conservative release category was selected.

2.2. Safe Shutdown Earthquake Event Tree

The first event tree is shown in Figure 1. The event tree models the occurrence of a SSE and the potential failure of the RPV supports. For this model, it is assumed that failure of the RPV supports may only cause a large break loss-of-coolant-accident (LBLOCA). Thus, a SBLOCA will not be considered as a contributor to the core-damage probability and will not be included in the event tree. This assumption is supported by the fact that the leak rate (less than 200 gpm) for a SBLOCA is easily replaced by various reactor makeup systems such as the high pressure safety injection system or the charging system.

Table 2. Event tree sequence end state categories.

Consequence Label	Explanation
SPRA	The sequence results in an event sequence whose risks are not associated with GSI-15. The sequence is not further developed on the event tree.
SF-PSD	The sequence results in an emergency plant shutdown. Thus, the plant is safe and in a shutdown mode.
PWR 1	The sequence results in core meltdown followed by a steam explosion. The containment sprays and heat removal systems are assumed to have failed. Radioactivity is released over a 10 minute period. The total release contains approximately 70% of the iodines and 40% of the alkali metals present in the core at the time of release.
PWR 3	The sequence results in containment failure prior to commencement of core melting. Core melting would cause radioactive materials to be released through a ruptured containment barrier. Approximately 20% of the iodines and 20% of the alkali metals present in the core at the time of release would be unleashed to the atmosphere. The release time would be approximately 1.5 hours.
PWR 7	The sequence results in core meltdown but is mitigated due to the fact that the containment barrier retains its integrity until the molten core melts through the containment. The release involves 0.002% of the iodines and 0.001% of the alkali metals present in the core at the time of release. The release time would be 10 hours.
PWR 8	The sequence results in large pipe break with failure of containment. The core would not melt. The release would involve 0.01% of the iodines and 0.05% of the alkali metals. Most of the release would occur in 0.5-hours.
PWR 9	The sequence results in a large pipe break. The core would not melt, and the containment would not fail. The release would contain 0.00001% of the iodines and 0.00006% of the alkali metals. The release would occur over a 0.5-hour time period.

The events for the SSE sequence event tree are defined below. The probability for each event is given as a mean value. The uncertainty analysis for the event trees is contained in Appendix A.

SSE. The event SSE models the initiating event of a safe shutdown earthquake. From NUREG-1211[3], it is assumed that frequency of occurrence is $2.5x10^{-4}$/RY and the initiating event probability can be modeled as a Poisson distribution. Thus, the dimensionless parameter ν for the Poisson distribution for one-year is $2.5x10^{-4}$.

It is commonly accepted that earthquakes producing lower loads on structures occur more often than those resulting in higher loads, such as the SSE. Also, if plastic design methods are used, as described in Chapter N of the AISC Manual[4], the allowable stress in the load combinations including SSE is $0.9xF_y$[5], i.e., for A36 steel this would result in about 30 ksi. Taking 6 ksi as the minimum stress corresponding to the threshold of embrittlement, the ratio of the stresses (the stress at the SSE and the minimum stress which might be considered for brittle fracture) would be equal to five. Since force is proportional to acceleration, and stress is proportional to force, it may be inferred that the stresses induced by an earthquake will be in the same proportion as peak ground acceleration (PGA) for different earthquakes.

Examination of the seismic hazard curves, relating annual frequency of exceedance of SSE and PGA[6], indicates that frequency of occurrence of the earthquake corresponding to 6 ksi would be about five times that of the frequency of occurrence of the PGA for the SSE. Reducing the stress threshold by a factor of five has the effect of increasing the frequency of the earthquake by about a factor of five. Consequently, it is justifiable to increase the dimensionless parameter ν for the Poisson distribution for one year by a factor of five, to $\nu = 1.25x10^{-3}$. The probability of at least one damaging earthquake is then:

$$P(SSE) = 1 - P(x=0) = 1 - \frac{\nu^x e^{-\nu}}{x!} = 1 - e^{-1.25x10^{-3}} = 1.25x10^{-3}$$

RPVSF. Event RPVSF represents the failure of the RPV supports if a damaging earthquake occurs. The calculation for the probability of RPVSF should be site specific due to variables such as RPV support design and material composition, plant age and operating history, and RPV load before and after the earthquake. In order to keep the analysis generic, the conditional probability of RPVSF is conservatively estimated to equal 0.5. This assumption implies that if a damaging earthquake occurs, fifty percent of the time the RPV supports will fail. Also, it implies that below the damaging earthquake level the RPV supports will not fail. In the sensitivity analysis contained in Appendix B, the frequency of having a damaging earthquake was increased by a factor of 10 to account for the possibility of a lower peak-ground acceleration level earthquake which results in RPV support failure.

LBLOCA. The event LBLOCA models a large break LOCA if the RPV support system undergoes a failure. If the RPV support does fail, the resulting load on the reactor cooling system (RCS) piping may cause a rupture. A conservative estimate of the probability of a LBLOCA is assumed to be 0.5. The sensitivity analysis in Appendix B investigated the worst case of RPV support failure coupled with a LBLOCA by setting both probabilities to 1.

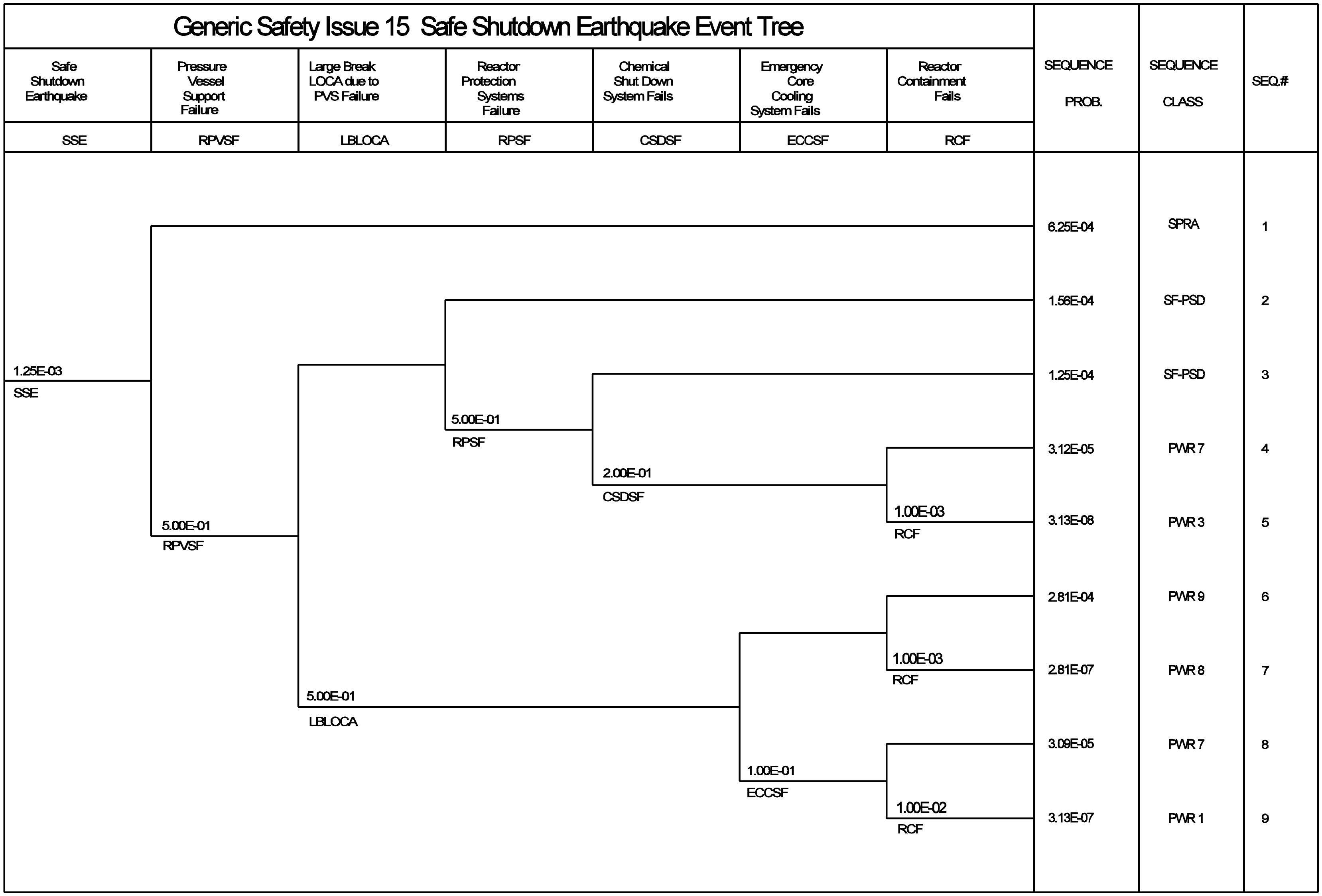

Figure 1. Safe Shutdown Earthquake Sequence Event Tree.

RPSF. The event RPSF models the failure of the reactor protection system. In the event of a SSE, the operator will attempt to manually scram the reactor. However, it is possible that the protection system will fail due to the tilting of the RPV, which causes the reactor control rods to become mechanically jammed. Thus, detailed analysis for this event should include both possible mechanical failures and human errors of commission.

Event RPSF is conditional on a LBLOCA not occurring. If a LBLOCA does occur, the moderator for the reactor will be removed and the reactor will shut down due to voiding of the core. Consequently, if a LBLOCA does not occur and the reactor protection system fails, the core will eventually melt even though the reactor coolant is still present. It is conservatively assumed that the probability of RPSF is 0.5. The sensitivity analysis in Appendix B investigated the worst case scenario by setting both RPVSF and RPSF to one. This would model coupled failure of both the RPV supports and the reactor protection system.

CSDSF. The event CSDSF models the failure of the chemical shutdown system. Typically, precise analysis of this event would include both the possible mechanical failures and human errors. For this analysis, the probability of CSDSF was found from the Sequoyah PRA[7] and is equal to 0.2. The sensitivity analysis in Appendix B investigated the worst case scenario by setting the probability of CSDSF failure to one.

ECCSF. Event ECCSF models the failure of the emergency core cooling system. If the emergency core cooling system works, it can prevent core melt even if a large break LOCA occurs. The probability of failure for this event is based upon typical PRA analysis. Based upon the Sequoyah PRA, the conditional probability of ECCSF is equal to 0.02. This mean conditional probability measures the failure probability of the systems (including human and mechanical) comprising the ECCS. Failure of ECCS following RPVS failure and the resulting RPV displacement and increased primary piping stresses would result in higher stresses on ECCS piping and components and would increase the conditional failure probability. Therefore, the event ECCSF probability was increased by a factor of five to 0.1. The sensitivity analysis in Appendix B investigated the worst case scenario (where the ECCS always fails given the appropriate initiating event) by setting the probability of CSDSF failure to one.

RCF. The event RCF represents the failure of the containment heat removal system along with the containment structure and containment isolation. Since the reactor containment and most of the systems in it have a median capacity of 1.5-2g peak ground acceleration, a SSE should not have a noticeable effect on the containment failure rate. Therefore, a typical PRA based failure rate is assumed for the reactor containment. Based upon the Sequoyah PRA, the probability of RCF is 1×10^{-3} in the mission time of one year. For those cases where ECCS has failed, the probability of RCF was assumed to be 1×10^{-2}, which accounts for the possibility of a dependent ECCS/RCF failure mode.

Also, the Salem Nuclear Generating Station PRA[8] was used to compare the failure rates of the mechanical components for the analysis of both event trees.

2.3. Small Break Loss-Of-Coolant Accident Event Tree

The second event tree is shown in Figure 2. The event tree models the occurrence on a non-seismic induced SBLOCA and the resulting failure of the RPV supports. Given that a SBLOCA occurs, the resulting load normally carried by the fractured pipes will be transferred to the RPVSs thereby causing an additional load on the supports. If the supports have undergone neutron embrittlement, the addition of the SBLOCA induced load may cause the RPVSs to fail. If the RPVSs do fail, a large break LOCA may occur.

As can be seen in Figure 2, the SBLOCA event tree is similar to that from the SSE sequence discussed above. The basis for using a similar event tree with identical event probabilities is that if the RPVSs fail, the possible resulting LBLOCA, reactor protection system failure, chemical shutdown system failure, emergency core cooling system failure, and reactor containment failure events will most likely fall within the same realm regardless of the cause of the RPVSs failing. Thus, the only difference between the two event trees is the initiating event and its associated frequency.

SBLOCA. The initiating event for the second event tree is the occurrence of a small break LOCA. The frequency of occurrence was obtained from the Sequoyah PRA[7] source numbers. The nominal frequency for a small break LOCA is found to be $1x10^{-3}$. The assumption was made that only one-half of the possible small pipe breaks were close enough to the RPV to load the RPV supports such that failure of the supports may occur. Therefore, the frequency of occurrence for a SBLOCA is estimated to be equal to the nominal SBLOCA frequency multiplied by one-half, or $5x10^{-4}$/RY. Assuming the SBLOCA event can be modeled as a Poisson event, the probability of a SBLOCA in the mission time of one year is equal to $5x10^{-4}$.

The remaining events in the SBLOCA event tree have previously been defined and will not be reviewed. In both the SSE and SBLOCA sequences the following preexisting conditions must be met before the RPVSs can fail:

- The support must contain a critically sized flaw.
- The support must have been subjected to enough radiation for embrittlement to occur.
- Sufficient stresses must be present to cause brittle fracture.

Requirements for toughness were implemented after some plants were built. Therefore, some older plants may have been at or near the NDT temperature at the beginning of plant life. If this is the case, they may be susceptible to brittle failures without significant exposure to neutron radiation.

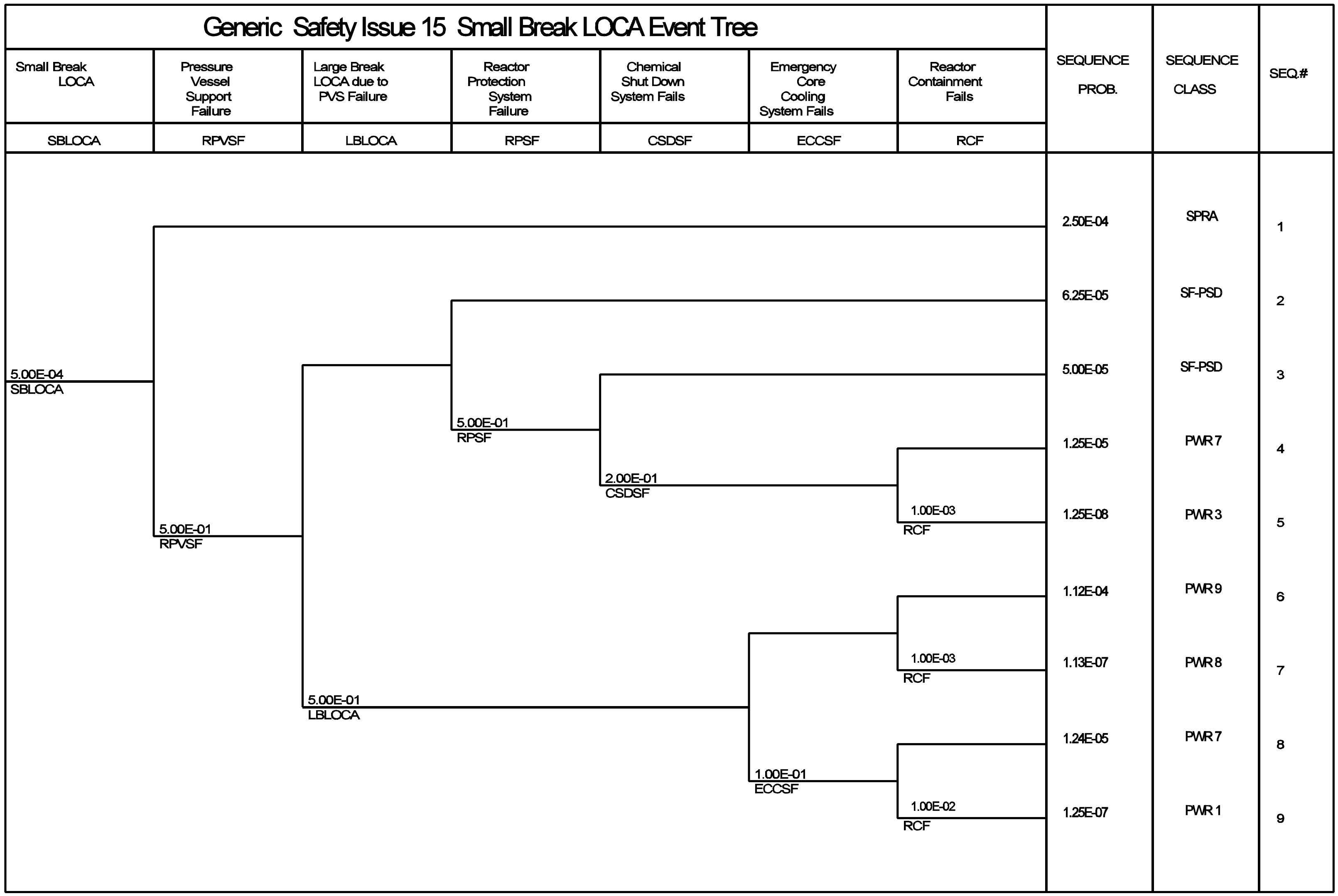

Figure 2. Small Break LOCA Event Tree.

2.4. Event Tree Results

Since the two event trees lead to similar sequence outcomes, the identical outcomes from each tree can be combined to form a total probability of a particular end state. The end states can occur with either the SSE or the SBLOCA as the initiator. Thus, the end state probability for the PWR 1 category is calculated by:

$$
\begin{aligned}
P(PWR\,1)_{total} &= P\left((PWR\,1)_{SSE} \cup (PWR\,1)_{SBLOCA}\right) \\
&\approx P(PWR\,1)_{SSE} + P(PWR\,1)_{SBLOCA} \qquad (if\ P(PWR\,1) << 1).
\end{aligned}
$$

Table 3 lists the sequence end states and the expected probability of occurrence for the mission time of one year. Since the expected probability is calculated by multiplying several random variables together, the probability distribution for the resultant product would tend to be log-normally distributed. As seen in Table 3, the total core damage frequency due to RPVS failure is $8.8x10^{-5}$/yr.

Table 3. Event tree end state analysis results.

Sequence end state	Expected probability (per year)	Core damage	Total cored damage frequency (per year)
SPRA	$8.8x10^{-4}$	No	N/A
SF-PSD	$3.9x10^{-4}$	No	
PWR 9	$3.9x10^{-4}$	No	
PWR 8	$3.9x10^{-7}$	No	
PWR 7	$8.7x10^{-5}$	Yes	$8.8x10^{-5}$
PWR 3	$4.4x10^{-8}$	Yes	
PWR 1	$4.4x10^{-7}$	Yes	

2.5. Sequence Risk Analysis

The risk is defined as the probability of occurrence of an event multiplied by the radioactive release consequence associated with the event. The risk is then extrapolated over the estimated remaining lifetime of a typical reactor. The risk from each event sequence is then summed to provide an upperbound total risk.

The fracture mechanics analysis reported in NUREG/CR-5320 described radiation embrittlement as a credible end-of-life failure mode, assuming 32 Effective Full Power Years (EFPY). Given the current "40-year license lifetime" and assuming an average plant is twenty years old, it is assumed that a remaining plant lifetime is 20 years, with the last 10 plant years encompassing the plausible radiation embrittlement failure mode.

Table 4 lists the consequence associated with each end state category. The consequence data quantify the WASH-1400 end states and are taken from NUREG/CR-2800[9]. The consequences for the SPRA and SF-PSD end states are both assumed to be zero (no additional risk).

Table 4. End state radioactive release consequences (from NUREG/CR-2800).

Category	Whole body dose consequence factor (person-rem)	
	Core melt	Non-core melt
PWR 1	5.4×10^6	
PWR 3	5.4×10^6	
PWR 7	2.3×10^3	
PWR 8		7.5×10^4
PWR 9		1.2×10^2

The WASH-1400 release categories were assigned to those event tree sequences that resulted in a radioactive release not covered by the normal plant specific design PRA. The release category that best fit each sequence was used to obtain an offsite dose for that sequence. As discussed in NUREG-2800, the total offsite radioactive dose was calculated based on the following assumptions:

1. Calculations were based on a typical midwest site, adjusted to reflect the population density within a 50-mile radius of U.S. nuclear power plants.

2. Dose consequences represent whole-body population dose commitment (person-rem) received within 50 miles of the site.

3. A 1/2-mile exclusion area was assumed, with a uniform population density of 340 persons per square mile from the exclusion area to the 50-mile exposure radius.

4. Evacuation was not considered.

5. Meteorological data were taken from the U.S. Weather Service station at Moline, Illinois.

6. Core inventory at accident initiation time was assumed to be represented by a 3412 MWt (1120 MWe) plant.

7. All exposure pathways except ingestion were included.

2.6. Risk Analysis Results

Table 5 lists the results of the risk analysis. The end state release consequence (Table 4) is multiplied by the end state probability (Table 3) to get an end state risk. The risk is then summed and multiplied by the remaining reactor lifetime to get the total additional population risk associated with the possible RPVS failure due to a SSE or a SBLOCA.

As shown in Table 5, the expected risk is 2.9 person-rem/year for the remaining plant lifetime after embrittlement occurs. The 2.9 person-rem/year risk is based on the operation of one reactor and is estimated using very conservative event probabilities. To get a total industry-wide risk value, the 2.9 person-rem/year should be multiplied by the total number of embrittlement susceptible plants and their respective remaining lifetimes. Assuming seventy-six susceptible plants, the total industry- wide risk value would be 2200 person-rem for a ten-year time period. If every embrittled support in the seventy-six plants were repaired, the expected total benefit from the reduction in risk would be 2200 person-rem.

Table 5. Risk analysis results.

Category	Expected risk (person-rem)
SPRA	0/year
SF-PSD	0/year
PWR 9	0.047/year
PWR 8	0.029/year
PWR 7	0.20 /year
PWR 3	0.24 /year
PWR 1	2.4 /year
$\sum =$	2.9 /year
x 10 years	29
x 20 years	58
x 40 years	120
x 60 years	170

3. COST EVALUATION

The proposed resolution modifications will have the effect of either preventing embrittlement from occurring, replacing or repairing potentially failed components, or changing the operating environment of embrittled components such that further embrittlement cannot occur. The risk reduction possible from the implementation of any of the proposed modifications is obtained from the event trees developed in Section 2.

3.1. Proposed Solution Options

Five possible options or alternatives were proposed as resolutions for GSI-15. It should be kept in mind that these are only potential solutions. A substantial engineering effort will be required before the feasibility of implementing any of these solutions at any given nuclear power plant is shown to be practical. The five options are:

1. Shielding the RPVSs from neutron radiation. This would prevent the RPVSs from becoming embrittled.

2. Increasing the operating temperature of the RPVSs above the new (embrittled) NDTT. This would remove the brittle fracture failure mode.

3. Replacing the existing RPVSs before embrittlement occurs.

4. Annealing the RPVSs to remove the effects of the embrittlement.

5. Strengthening the existing RPVSs or adding new supports.

3.2. Discussion Of Options

Before any proposed modification could be made to resolve this issue, an extensive engineering analysis would be required on a plant-by-plant basis. Included in this effort, the analysis would have to: assess the effects of neutron embrittlement on a plant-specific basis, calculate the risk associated with the possible embrittlement, insure that the implementation of any proposed modification is possible and that it will actually solve the problem, perform the design and engineering work for any proposed modification, pass the required engineering reviews, and obtain NRC design approval.

It should be noted that the RPVSs are located in an area of high radiation with extremely limited access. Even the act of visually inspecting them would be a major undertaking, which would result in a considerable occupational exposure. Therefore, any proposed solution needs to be evaluated both on the merits of its cost-to-benefit ratio and in light of the additional occupational exposure that would result from its implementation.

Option 1 is to shield the RPVSs from neutron radiation. This would prevent the RPVSs from becoming embrittled. Because of the limited space available in the area of the RPVSs, adding shielding would not be practical unless a shielding with an extremely large neutron absorption cross-section is used. The procurement of suitable shielding would probably be expensive. Also, the shielding must not interfere with the normal, inherent heat transfer mechanisms of the RPVSs.

Option 2 is to increase the operating temperature of the RPVSs above the new (embrittled) NDTT. This would remove the brittle fracture failure mode. It is questionable if this option is applicable to the short column RPVSs. The short column supports have a small profile with a large temperature differential. In NUREG/CR-5320 it is estimated that for the Trojan plant, after 32 EFPY, there will be a 75°F shift in the NDTT in the area most likely to contain a critically-sized flaw. In order to elevate the RPVS's operating temperature sufficiently to accommodate this shift, it would probably require exceeding the temperature limit of the supporting concrete. This would have the effect of changing the failure mechanism from failure of the RPVS to failure of the supporting concrete structure.

Option 3 is to replace the existing RPVSs before embrittlement occurs. It is unlikely that this option could be completed during a scheduled shutdown. It would, therefore, involve the buying of replacement power. Because the RPVSs are keyed to the RPV nozzles, the replacement of the RPVSs would most likely involve either lifting or removing the RPV until the supports are replaced.

Option 4 is to anneal the RPVSs to remove the effects of neutron induced embrittlement. There are two methods by which the RPVSs could be annealed. The possibilities are either in-place annealing of the supports or removing the supports and annealing at a remote location. In-place annealing would probably be the most cost effective; however, for those RPVSs that are attached to or imbedded in concrete (i.e. short column RPVSs) this may not be possible due to the temperature limit of the supporting concrete. Option 4 is calculated in two ways; the first way (Option 4A) takes into account removal of the RPVSs to an out-of-containment location for annealing, and the second way (Option 4B) calculates the cost of in-place annealing. Like Option 3, both options would most likely involve the buying of replacement power.

Option 5 is to strengthen the existing RPVSs or add new supports. It is questionable whether or not this option is possible. For most reactors, all the locations that can be used to support the RPV are currently in use, and any attempt to strengthen the existing supports would be akin to replacing the RPVSs, with all the implementation problems associated with Option 3. Like Options 3 and 4, this option would involve the buying of replacement power during the modification downtime.

3.3. Cost Analysis Methodology

The cost estimates of the five options were developed using the guidelines of NUREG/CR-3568[10], "A Handbook for Value-Impact Assessment," and NUREG/CR-4627[11], Revision 2, "Generic Cost Estimates," and the computer code FORECAST 2.1[12], which incorporates the cost evaluation information. FORECAST was developed under the auspices of the NRC. It has been used as the basis for estimating costs in several cost/benefit analyses prepared for the NRC. Cost estimation involved making an evaluation of each proposed modification, identifying equipment and materials necessary to make the proposed modifications, and assessing the work area in which the proposed modifications would be made. The following assumptions were included in the cost estimates:

1. If implemented, the solution would resolve the problem with 100% assurance.

2. Options 3, 4A, and 5 probably cannot be implemented without replacement power costs. Options 1, 2, and 4B may possibly be implemented without buying replacement power.

3. Socio-economic impacts will be considered minimal and will not be included as an increment of cost.

4. Costs were calculated using 1991 dollars.

5. Costs were calculated assuming that modifications would be required on the total support system.

6. For Option 1, shielding would have to be constructed from an alloy of cadmium. Based upon engineering judgement, material costs would be approximately $50,000 per support, for a total cost of $200,000 for four supports.

7. Option 4 has no equipment or materials costs.

8. Options 1, 2, and 5 have no removal labor costs associated with them. Option 4B removal costs would be the cost associated with removal of the annealing equipment and is estimated to be one-third of the installation cost.

9. Due to the high radiation dose present in the area containing the RPVSs, no modifications could be made without first defueling and draining the reactor vessel.

10. For Option 2, NUREG-0933[13] estimated that some plants would have material costs as low as $5200 and labor costs as low as $25,000. The numbers were calculated based on the assumption that the temperature of the RPVSs could be raised above the new NDTT by simply adjusting cooling flow to the RPVSs. We feel that even if this fix is possible, it would require the installation of additional temperature monitoring equipment, such that the cost would be similar to the costs associated with the installation of heating systems discussed in NUREG-0933.

11. The cost of buying replacement power was made on the assumption that Options 1, 2, and 4B would require an additional 4 weeks of outage time, Options 3 and 4A would require an additional 20 weeks, and Option 5 would require an additional 16 weeks.

Expenses were calculated in accordance with FORECAST 2.1. The total cost of a modification is the sum of many different types of expenditures. The costs that were analyzed were limited to the following categories:

1. Equipment and material costs.

2. Labor costs associated with installation and/or removal.

3. Costs associated with engineering and quality control and quality assurance (QA/QC).

4. Radiation exposure.

5. Costs associated with health physics.

6. The costs to defuel, drain, and restore the reactor.

7. Replacement power costs.

8. Total NRC costs, both one-time and recurring costs.

9. Averted onsite costs (AOSC).

3.4. Cost Estimate Categories

Labor, Equipment, and Material Costs

The Energy Economic Data Base (EEDB), which is built into the FORECAST code, provided the basis for the equipment costs, material costs, and labor estimates. The EEDB incorporates "as-built" cost information (both the material unit cost and the installation or removal labor hours) for nuclear plant activities. Additionally, for operating nuclear power plants there are a number of workplace characteristics which significantly reduce the level of productivity and thus increase the number of labor hours required to accomplished a task. These characteristics, discussed in detail in FORECAST 2.1, include access, congestion and interference, radiation, task management, etc. Since the EEDB reflects only new (or "as-built") plant conditions, the installation labor hours were adjusted using FORECAST 2.1 to properly consider actual working conditions existing at operating nuclear plants. FORECAST 2.1 can modify the EEDB to take into account the factors that reduce worker productivity.

The total labor costs associated with the proposed modifications include overhead charges to account for contractor management, administrative support, rent, insurance, etc. Options 1, 2, and 4B installation labor hours were estimated based on 105 man-weeks obtained from Reference 13. The labor hours and material costs associated with Option 3 were obtained directly from the EEDB. Option 4A labor hours were assumed to be the same as Option 3, but its material costs were assumed to be zero. Option 5 material costs were assumed to be the same as Option 3, but the labor hours were adjusted to reflect that there would be no removal costs associated with Option 5.

Costs Associated with Engineering and QA/QC

These costs reflect the cost of engineering and design, as well as quality assurance and quality control (QA/QC) activities associated with implementing the requirements. For requirements affecting structures or systems already in-place (operating plants), the guidelines of Abstract 6.4 of FORECAST recommend a 25% engineering and QA/QC factor be applied to the direct cost (i.e., the labor and materials cost without any overhead charges). All cost estimates developed in this study include this engineering and QA/QC cost component. In the case of Options 1, 2, and 4B, a large analytical effort would be required to insure that the implementation of any proposed modification is possible, that it will actually solve the problem, and that it can acquire NRC design approval. Therefore, for these two options a 40% engineering and QA/QC factor was applied.

Radiation Exposure Estimation

Worker radiation exposure estimates were derived based on guidelines presented in FORECAST. The collective radiation exposure associated with the implementation of a proposed plant modification is estimated by taking the product of the in-field labor hours necessary to perform the task and the work area dose rate associated with that particular task.

In this study, the work area in which the modifications would take place is considered to be high-dose contaminated area (inside the biological shield). Based on engineering judgement, radiation exposure level (with the reactor's fuel removed) is estimated to be 10 mrem/hour for the proposed modifications.

Costs Associated with Health Physics

Health physics requirements for the potential plant modifications were developed based on information and guidelines presented in Abstract 2.1.6 of Reference 12. Two factors were considered: the size of the work crew and the magnitude of the radiation field. The plant health physicist (HPs) monitor personnel radiation doses, perform radiological surveys throughout the modification duration, staff radiological checkpoints, set up anti-contamination clothing removal areas, as well as determine allowable stay times and badging requirements.

Cost to Defuel, Drain, and Restore the Reactor

If the nuclear reactor core is left in place, high radiation levels (2-3 REM/hr)[a] would be experienced in the area where the modifications would be made. Therefore, if any modification is to be made, the reactor must be defueled and drained and then refueled after the modifications are completed. In accordance with Abstract 2.1.3 of Reference 12, these defueling and restoring costs were developed for a typical PWR. Not included in these costs are the costs associated with fuel sipping and vessel surveillance and inspection.

Replacement Power Costs

Replacement power costs for the potential plant modifications were developed based on information and guidelines presented in Abstract 2.1.2 of Reference 12. A best estimate of $500,000/day was used, with high and low values of $900,000/day and $150,000/day, respectively.

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[a] As measured in the area of the reactor vessel nozzle's at the Trojan Nuclear Power Plant, per telephone conversation between R.W. Garner of the INEL and Arnie Fero of Westinghouse Electric on 5/16/90.

Total NRC Costs

The total NRC costs include the one-time cost associated with supporting the implementation of any proposed modifications and the recurring costs associated with reviewing the operation and maintenance of a modification after it is implemented.

NUREG-2800 estimated it would take 16 man-weeks of staff effort to develop possible solutions. At a rate of $45.35 per hour, this amounts to $29,000. Supplementary contractor support was estimated to cost an additional $500,000, for a total cost of $529,000 for all 76 affected plants (or $6960/plant).

NRC efforts to support and review implementation of any modification was estimated by NUREG-2800 to be 15 man-weeks/plant. Also, it was estimated that for some modifications only 2 man-weeks would be required. However, due to the complicated issues involved in all of the proposed modifications, we feel the 15 week figure applies to all modifications. At a rate of $45.35 per hour, the 15 man-weeks/plant totals $27,000 per plant.

Recurring costs were estimated to be 1 man-week/RY per plant. Given ten years of remaining reactor life, at a cost of $45.35 per hour, this amounts to $18,100 per plant. Based on the above estimates, the total NRC cost per plant is given by:

$(6960 + 27,000 + 18,100) = $52,000

Averted Onsite Costs (AOSC)

In addition to the costs associated with the modification, the potential reduction of severe onsite consequences was evaluated. A Handbook for Value-Impact Assessment was used as the reference for this evaluation. The AOSC was calculated using the following equation:

$$V_{op} = NU(F_O - F_N)$$

where

V_{op}	=	the cost of avoided onsite property damage
N	=	the number of affected facilities (on a per plant basis, N = 1)
U	=	the present value of onsite property damage given a release
F_O	=	the original core damage frequency (base case)
F_N	=	the core damage frequency after implementing an option (assumed to be zero)
F_O-F_N	=	8.8E-5 (from summation of core melt frequencies contained in Table 3)

and

$$U = \frac{C}{m}\left(\frac{e^{-rt_i}}{r^2}\right)\left(1-e^{-r(t_f-t_i)}\right)\left(1-e^{-rm}\right)$$

where

C	=	cleanup, repair, and replacement power costs ($\$1.65x10^9$, the data associated with scenario 3)
t_f	=	years remaining until end of plant life (10 years)
t_i	=	years before reactor begins operation (0 years)
m	=	period of time over which damage costs are paid out (10 years)
r	=	discount rate (for 10%, r = 0.10).

When uncertainty in the calculation of V_{op} is considered, it is appropriate to calculate a low, best, and high estimate for the value of U. These values can then be multiplied by the change in core damage frequency to yield a low, best, and high value for V_{op}. The cost handbook was used as a guide, and the best, high, and low estimate values for U were determined by:

1. The best estimate was calculated as discussed above.

2. The high estimate was assumed to be three times the best estimate.

3. The low estimate was calculated using data from scenario 2 ($103.5M over 7.5 years).

<u>A Handbook for Value-Impact Assessment</u> states that "the quantity, U, must be interpreted carefully to avoid misunderstandings. It does not represent the expected onsite property damage due to a single accident, Rather, it is the present value of a stream of <u>potential</u> losses extending over the remaining lifetime of the reactor. Thus, it reflects the expected loss due to a single accident, the possibility that such an accident could occur with some small probability at any time over the remaining reactor life, and the effects of discounting these potential future losses to present value. When the quantity, U, is multiplied by the accident frequency, the result is the expected loss over the reactor life, discounted to present value."

The best, high, and low present onsite property damage costs (including cleanup cost, repair and refurbishment cost, and replacement energy cost) given a release were calculated as:

Low estimate of U = $\$4.6x10^8$ /severe accident event
Best estimate of U = $\$6.6x10^9$ /severe accident event
High estimate of U = $\$2.0x10^{10}$/severe accident event

These values were then applied to the potential change in accident frequency to obtain dollar values for AOSC, as follows:

V_{op}(Low Estimate) = $40,500[a]
V_{op}(Best Estimate) = $581,000[a]
V_{op}(High Estimate) = $1,760,000[a]

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[a] V_{op} is dependent on the remaining plant lifetime (t_f). These values were obtained using a 10-year t_f. If the remaining plant lifetime increases to 60 years, the best estimate of V_{op} increases to $922,000. This will not have a significant impact on the cost/benefit results. Therefore, only the 10-year remaining lifetime AOSC value was used.

3.5 Cost Evaluation Uncertainty

The areas of uncertainty associated with the cost estimating model for this study included the following:

1. Labor rate variations due to plant site location,

2. Variability of in-plant work environment conditions,

3. Variations in the cost of replacement power,

4. NRC procedural/administrative/analytical cost,

5. Equipment and material costs variations,

6. The degree of engineering effort required to obtain NRC approval of any proposed modification.

Each proposed option's cost estimate was evaluated to determine the areas of uncertainty. For the cost analysis uncertainty, the following assumptions were made:

1. Labor rate variations due to plant site location are considered when calculating labor costs. In accordance with FORECAST recommendation for labor cost variations, the assumed labor rate variation was as follows: best estimate is 100% of the labor cost, the high cost estimate is 112%, and the low cost estimate is 88%. These variations are applicable to installation and removal labor, health physics labor, NRC labor, and the costs associated with defueling the reactor.

2. Equipment and material costs were obtained from the FORECAST data base (or, in the case of Option 2, from NUREG-2800). The low estimate was assumed to be 75% of the best estimate and the high value was assumed to be 125% of the best estimate.

3. Best estimates for engineering QA-QC costs were obtained using FORECAST. However, due to the large uncertainty in the degree of the engineering effort required to obtain NRC approval of any proposed modification, the low estimate was assumed to be 50% of the best estimate and the high estimate was assumed to be 150% of the best estimate.

4. Cost estimates for buying replacement power were found from the FORECAST data base. A best estimate of $500,000/day was used, with high and low estimates of $900,000/day and $150,000/day, respectively.

Table 6 shows the mean, the coefficient-of-variation (COV), and the standard deviation of each cost category for the five different proposed modifications. The COV is defined as the standard deviation divided by the mean and is a measure of the possible variation in the cost. For a detailed discussion of uncertainty calculations, see Appendix A.

3.6. Plant Modification Cost Estimate Results

A mean and standard deviation for the total cost of each option was calculated for each modification by using a numerical Taylor series expansion routine. Table 7 lists the cost results for the various modifications. Included in the table are the total cost estimate without AOSC or replacement power, the total cost including AOSC without replacement power, and the total cost with both AOSC and replacement power.

It should be noted that the normal costs are considered to be positive dollars. The AOSC cost is measured in negative dollars, thereby helping to lower the total costs.

Table 6. Cost analysis category parameters.

Cost Category	COV	Option[a] 1		Option 2		Option 3		Option 4A		Option 4B		Option 5	
		Mean ($)	Std. dev.	Mean ($)	Std. dev.	Mean ($)	Std. dev.	Mean ($)	Std. dev.	Mean ($)	Std. dev.	Mean ($)	Std. dev.
Equipment and Materials	25%	200K[b]	50K	52K	13K	1M	250K	n/a	n/a	n/a	n/a	1M	250K
Installation Labor	12%	770K	92K	770K	92K	10M	1.2M	10M	1.2M	770K	92K	10M	1.2M
Removal Labor	12%	n/a	n/a	n/a	n/a	3.4M	408K	3.4M	408K	250K	30K	n/a	n/a
Engineering QA/QC	50%	390K	195K	330K	165K	1.9M	950K	1.8M	900K	410KM	205K	1.5M	750K
Health Physics	12%	150K	18K	150K	18K	3.1M	372K	3.1M	372K	200K	24K	2.5M	300K
Defuel, Drain, and Recover	12%	165K	20K	165K	20K	165K	20K	165K	20K	165K	20K	165K	20K
Replacement Power	c	14M	3.5M	14M	3.5M	70M	18M	70M	18M	14M	3.5M	56M	14M
NRC Cost	12%	52K	6.2K	52K	6.2K	52K	6.2K	52K	6.2K	52K	6.2K	52K	6.2K
AOSC (-$)	c	581K	226K	581K	226K	581K	226K	581K	226K	581K	226K	581K	226K

[a] For a description of the different options refer to Section 3.1.
[b] K = thousand, M = million.
[c] Standard deviation is found by $3\sigma = ([High\ value - Best] + [Best - Low\ value])/2$, where σ = standard deviation.

Table 7. Cost analysis results.

Cost type	Option 1		Option 2		Option 3		Option 4A		Option 4B		Option 5	
	Mean ($)	Std. Dev.	Mean ($)	Std. Dev.	Mean ($)	Std. Dev.	Mean ($)	Std. Dev.	Mean ($)	Std. Dev.	Mean ($)	Std. Dev.
Total Cost w/o AOSC & w/o RP[a]	1.7M[b]	220K	1.5M	190K	20M	1.6M	19M	1.6M	1.8M	230K	15M	1.5M
Total Cost w/o RP	1.1M	320K	920K	300K	19M	1.6M	18M	1.6M	1.3M	320K	15M	1.5M
Total Cost	15M	3.5M	15M	3.5M	89M	18M	88M	18M	16M	3.5M	71M	14M

[a] RP = replacement power.
[b] K = thousand M = million.

3.7. Radiation Exposure

The occupational radiation exposure results are presented in Table 8. These doses were calculated based on a 10 mrem/hour radiation field. This dose rate was applied only to those installation or removal labor hours that were estimated to be performed in the radiation area (37.5% of total installation or removal labor hours). Due to the congested nature of the area where the work would be performed, the installation of additional shielding to lower the exposure would not be possible.

Table 8. Total occupational radiation exposure.

Exposure	Option 1	Option 2	Option 3	Option 4A	Option 4B	Option 5
Total labor hours	4,200[a]	4,200[a]	90,000[b]	90,000[b]	5,600[b]	71,000[b]
Labor hours in radiation zone	1,600	1,600	33,000	33,000	21,000	25,000
Total exposure (person rem)	16	16	330	330	21	250

[a] Estimated from Reference 13.
[b] Estimated from FORECAST data base.

The total exposures presented in Table 8 represent the total dose that would be received by the labor force. This total dose would be distributed throughout the work force performing the implementation of an option. The site as-low-as-reasonable-achievable (ALARA) program should ensure that none of the individual workers exceeds the maximum dose rates set by 10 CFR Part 20.

4. COST/BENEFIT ASSESSMENT

4.1. Dollar-to-Person-Rem Averted Ratio

One measure of the benefit achieved by modifying a plant, is the Dollar-to-Person-Rem Averted Ratio (DPR) as described in Reference 11. A value of $1000 per person-rem is generally used by the NRC as an upperbound guideline in deciding whether corrective measures may be appropriate. The DPR is calculated as the modification cost divided by the offsite person-rem averted if the modification is performed, or:

$$DPR = \frac{Modification\ Cost}{Averted\ Offsite\ Dose}$$

NRC policy recommends inclusion of the AOSC in the expression for the DPR. The inclusion of averted onsite costs reduces the cost of the modification, causing the cost benefit ratio to becomes more favorable. The DPR could then be calculated by:

$$DPR = \frac{Modification\ Cost - AOSC}{Averted\ Offsite\ Dose} = \frac{Total\ Costs}{Averted\ Offsite\ Dose}$$

4.2. Cost/Benefit Results

The results of the cost/benefit analysis were calculated using the formulas presented above, the modification costs developed in Section 3, and the offsite doses developed in Section 2. Tables 9 through 14 show the cost/benefit results for the GSI-15 modifications (options 1-5), including the case where the occupational exposure is included in the calculation. Inclusion of the occupational dose is accomplished by subtracting the occupational exposure from the averted offsite dose, or:

$$DPR = \frac{Total\ Costs}{Averted\ Offsite\ Dose - Occupational\ Dose}$$

For those cases where the occupational exposure exceeds the averted offsite dose, no net benefit (NNB) is reported as the result. This is done because once the benefit becomes zero or less, the cost/benefit ratio indicates that performing the modification will result in a larger occupational dose than what would be expected for the populational dose if the modification is *not* implemented.

Tables 9 through 14 include the best estimates for 10 year, 20 year, 40 year, and 60 year remaining lifespans (see Appendix A and Appendix C for an example of the uncertainty calculations). The remaining lifespan is the time left to operate the plant *after* the supports have become brittle. The results in the tables are calculated for the three cost categories: without either AOSC or replacement power, with AOSC but without replacement power, and with both AOSC and replacement power. The calculated values are considered to be best estimate values. Graphical results are presented in Appendix

D. The graphs are given to assist in evaluating the relative cost/benefit magnitudes between the different options and the different cost categories.

Table 9. Cost/Benefit results for Option 1.

Years after embrittlement	Cost/Benefit (without occupational dose) [$/person-rem]			Cost/Benefit (with occupational dose) [$/person-rem]		
	Total w/o AOSC & w/o RP[a]	Total w/o RP	Total	Total w/o AOSC & w/o RP	Total w/o RP	Total
10	59K[b]	38K	520K	130K	86K	1.2M
20	30K	19K	260K	41K	26K	360K
40	15K	9.5K	130K	17K	11K	150K
60	9.8K	6.3K	87K	11K	7.0K	96K

[a] RP = Replacement Power
[b] K = thousand, M = million

Table 10. Cost/Benefit results for Option 2.

Years after embrittlement	Cost/Benefit (without occupational dose) [$/person-rem]			Cost/Benefit (with occupational dose) [$/person-rem]		
	Total w/o AOSC & w/o RP[a]	Total w/o RP	Total	Total w/o AOSC & w/o RP	Total w/o RP	Total
10	52K[b]	32K	520K	120K	72K	1.2M
20	26K	16K	260K	36K	22K	360K
40	13K	8K	130K	15K	9.3K	150K
60	8.7K	5.3K	87K	9.6K	5.9K	96K

[a] RP = Replacement Power
[b] K = thousand, M = million

Table 11. Cost/Benefit results for Option 3.

Years after embrittlement	Cost/Benefit (without occupational dose) [$/person-rem]			Cost/Benefit (with occupational dose) [$/person-rem]		
	Total w/o AOSC & w/o RP[a]	Total w/o RP	Total	Total w/o AOSC & w/o RP	Total w/o RP	Total
10	690K[b]	660K	3.1M	NNB[c]	NNB	NNB
20	350K	330K	1.6M	NNB	NNB	NNB
40	170K	170K	780K	NNB	NNB	NNB
60	120K	110K	520K	NNB	NNB	NNB

[a] RP = Replacement Power
[b] K = thousand, M = million
[c] NNB = no net benefit

Table 12. Cost/Benefit results for Option 4A.

Years after embrittlement	Cost/Benefit (without occupational dose) [$/person-rem]			Cost/Benefit (with occupational dose) [$/person-rem]		
	Total w/o AOSC & w/o RP[a]	Total w/o RP	Total	Total w/o AOSC & w/o RP	Total w/o RP	Total
10	660K[b]	630K	3.1M	NNB[c]	NNB	NNB
20	330K	320K	1.6M	NNB	NNB	NNB
40	170K	160K	780K	NNB	NNB	NNB
60	110K	110K	520K	NNB	NNB	NNB

[a] RP = Replacement Power
[b] K = thousand, M = million
[c] NNB = no net benefit

Table 13. Cost/Benefit results for Option 4B.

Years after embrittlement	Cost/Benefit (without occupational dose) [$/person-rem]			Cost/Benefit (with occupational dose) [$/person-rem]		
	Total w/o AOSC & w/o RP[a]	Total w/o RP	Total	Total w/o AOSC & w/o RP	Total w/o RP	Total
10	63K[b]	45K	560K	230K	170K	2.1M
20	32K	23K	280K	49K	36K	440K
40	16K	11K	140K	19K	14K	170K
60	11K	7.5K	93K	12K	8.6K	110K

[a] RP = Replacement Power
[b] K = thousand, M = million

Table 14. Cost/Benefit results for Option 5.

Years after embrittlement	Cost/Benefit (without occupational dose)			Cost/Benefit (with occupational dose)		
	Total w/o AOSC & w/o RP[a]	Total w/o RP	Total	Total w/o AOSC & w/o RP	Total w/o RP	Total
10	520K	520K	2.5M	NNB[c]	NNB	NNB
20	260K	260K	1.3M	NNB	NNB	NNB
40	130K	130K	630K	NNB	NNB	NNB
60	87K	87K	420K	NNB	NNB	NNB

[a] RP = Replacement Power
[b] K = thousand, M = million
[c] NNB = no net benefit

5. SUMMARY OF COST/BENEFIT FINDINGS

The cost results (see Table 7) indicate the estimated per plant costs range from a low value of \$920,000 for Option 2 (increasing the operating temperature of the supports) to a high value of \$89,000,000 for Option 3 (replacing the existing supports). The low value takes into account averted onsite costs and assumes no need to purchase replacement power. The high value also takes into account averted onsite costs, but assumes replacement power would have to be purchased for a 20-week period.

The results of the benefit analysis indicate a per plant offsite dose risk of 2.9 person-rem/year with a calculated core damage frequency of 8.8×10^{-5}/yr. The risk value includes all the risk associated with support failure after embrittlement occurs. It was assumed that the implementation of any of the proposed options would remove 100% of the risk associated with failure of an embrittled support.

The above information provided best estimate cost/benefit ratios ranging from \$5,300 per person-rem (Option 2 with AOSC and without replacement power and occupational dose over a 60-year embrittlement period) to \$3,100,000 per person-rem (Options 3 and 4 with AOSC and replacement power and without occupational dose over a ten year embrittlement period). When the occupational dose is considered, the cost benefit ratios increase. In those cases where the occupational dose exceeds the averted offsite dose, no net benefit is obtained.

Appendix B presents a number of sensitivity studies to show how the results can change given changes in the modeling data. Table B-2 gives four extreme cases of cost/benefit. It should be pointed out that these extreme cases represent the worst possible case for the cost/benefit analysis. In the case of minimum cost/maximum benefit, a potential cost/benefit ratio of \$53 per person-rem is obtained. This represents the case where the minimum-cost option would correct the problem for a plant located in an area of high populational density (assuming no occupational dose and a 60-year embrittlement period).

6. REFERENCES

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APPENDIX A - GSI-15 Event Tree Uncertainty Analysis

The event tree uncertainty analysis was initiated by assigning an appropriate uncertainty to each event in both of the event tree sequences. Table A.1 lists each event with its mean value, standard deviation, and assumed underlying probability distribution type. The source listed in the table is the source of the event *mean* value. The standard deviation value for the two Poisson initiating events were calculated by the relationship of: standard deviation = $\sqrt{mean}$.[a] The standard deviation value for the log-normal distributions in the table were estimated based upon engineering judgement.

Table A.1. Sequence Event Uncertainty Parameters.

Event	Mean	Standard deviation	Distribution type	Source
SSE	1.25×10^{-3}	3.5×10^{-2}	Poisson	Ref. 3
SBLOCA	5.0×10^{-4}	2.2×10^{-2}	Poisson	Ref. 7
RPVSF	5.0×10^{-1}	2.0×10^{-1}	Log-normal	EJ[b]
LBLOCA	5.0×10^{-1}	2.0×10^{-1}	Log-normal	EJ
RPSF	5.0×10^{-1}	2.0×10^{-1}	Log-normal	EJ
CSDSF	2.0×10^{-1}	1.0×10^{-1}	Log-normal	Ref. 7
ECCSF	1.0×10^{-1}	1.0×10^{-1}	Log-normal	EJ
RCF	1.0×10^{-3}	5.0×10^{-3}	Log-normal	Ref. 7
	1.0×10^{-2}	5.0×10^{-2}		EJ

[b] EJ = Engineering Judgement.

Normally probabilistic risk assessments assign log-normal distributions to the individual events contained in event trees. This arbitrary assignment of distributions stems from the fact that the log-normal distribution efficiently models events with low probabilities. But, for unlikely events (such as an earthquake) that occur at a constant rate and that change the system once the event does occur, a Poisson distribution is frequently used as the underlying distribution[b].

[a] This relationship is only valid on Poisson distributed events.

[b] The PRA Procedures Guide, USNRC Report NUREG/CR-2300, Jan. 1983, illustrates calculating the occurrence of earthquakes by using the Poisson distribution. Other probability and statistics texts and seismic reports verify that events such as an earthquake may be modeled by the Poisson distribution.

In Table A.1, event RCF is listed as having two parameters. The first parameter ($1.0x10^{-3}$) models the normal, independent failure of the containment, while the second parameter ($1.0x10^{-2}$) models the correlated failure mode of the emergency core cooling system and the reactor containment.

The event tree sequences were analyzed using a numerical Taylor series expansion routine to find the mean and standard deviation for each sequence outcome. The Taylor series expansion program was written by one of the authors (Smith) and was verified, both by hand calculations and textbook problems, before use on this project. Appendix C presents two samples of the program verification.

Table A.2 lists the sequence end states expected probability, 95th percentile probability, and standard deviation. The probability distribution for each sequence outcome is assumed to be log-normally distributed due to the multiplication of several events. The expected probability and standard deviation were obtained from the Taylor series expansion program. The 95th percentile valve was calculated using the obtained expected value and standard deviation and the assumption that the resulting distribution was log-normal.

Table A.2. Event tree sequence end state results.

Sequence end state	Mean probability (per year)	95th percentile (per year)	Standard deviation (per year)
SPRA	$8.8x10^{-4}$	$2.3x10^{-3}$	$2.1x10^{-2}$
SF-PSD	$3.9x10^{-4}$	$1.2x10^{-3}$	$6.6x10^{-3}$
PWR 9	$3.9x10^{-4}$	$1.0x10^{-3}$	$9.3x10^{-3}$
PWR 8	$3.9x10^{-7}$	$1.0x10^{-6}$	$9.4x10^{-6}$
PWR 7	$8.7x10^{-5}$	$2.6x10^{-4}$	$1.5x10^{-3}$
PWR 3	$4.4x10^{-8}$	$1.2x10^{-7}$	$1.0x10^{-6}$
PWR 1	$4.4x10^{-7}$	$1.2x10^{-6}$	$1.0x10^{-5}$

Table A.2 lists the 95th percentile values for the sequence end state distribution. The different percentile values (5th, 50th, and 95th) and error factor (EF) for a log-normal distribution are calculated using the equations below. Traditionally, the 5th percentile is considered a lower bound while the 95th percentile is an upper bound.

$$EF = e^{1.645\left(\ln[1+(\sigma/\mu)^2]\right)^{1/2}}$$

$$median = 50th = \frac{\mu}{[1+(\sigma/\mu)^2]^{1/2}}$$

$$95th = median \cdot EF$$

$$5th = \frac{median}{EF}$$

where

σ = log-normal standard deviation

μ = log-normal mean

The risk is defined as the probability of an event multiplied by the release consequence of the event. The risk is then extrapolated over the estimated remaining lifetime of a typical reactor. Most of the embrittlement of the RPVSs occur early in the lifetime of a plant. For the purpose of illustration in this appendix, the analysis assumes that the plant has a 10 year remaining lifetime. The risk from each event sequence is then summed for the 10 years to get an upperbound total risk.

Table A.3 lists the whole body dose consequence associated with each end state category. The consequence data quantifies the WASH-1400 end states and is taken from NUREG/CR-2800. The consequence for the SPRA and SF-PSD end state are both assumed to be zero (no additional risk). The consequence dose values are not treated as uncertain variables. Rather, the values are handled as upper bound numbers, which requires the values to be treated as conservative point estimates.

Table A.3. End state radioactive release consequences.

Category	Consequence factor (person-rem)	
	Core Melt	Non Core Melt
PWR 1	$5.4x10^6$	
PWR 3	$5.4x10^6$	
PWR 7	$2.3x10^3$	
PWR 8		$7.5x10^4$
PWR 9		$1.2x10^2$

Table A.4 lists the results of the risk analysis. The end state release consequence is multiplied by the end state probability to get an end state risk. The risk is then summed and multiplied by the 10-year duration to get the total additional population risk associated with the possible RPV support failure due to a SSE or a SBLOCA.

Table A.4. Risk analysis uncertainty results.

Category	Expected risk (person-rem)	Standard deviation (person-rem)	95th Percentile risk (person-rem)
SPRA	0/year	0/year	0/year
SF-PSD	0/year	0/year	0/year
PWR 9	0.047/year	1.12/year	0.12 /year
PWR 8	0.029/year	0.71/year	0.077/year
PWR 7	0.20 /year	3.5/year	0.58 /year
PWR 3	0.24 /year	5.4/year	0.65 /year
PWR 1	2.4 /year	54.0 /year	6.4 /year
$\sum =$	2.9 /year	54 /year	8.2 /year
x 10 years	29	540	82

Table A.4 shows the expected risk is 29 person-rem for the entire ten year embrittlement duration. Accounting for the uncertainties in the event tree analysis gives a 95th percentile risk of 82 person-rem.

Figure A.1 shows the cumulative probability distribution curve for the base case risk. The base case median risk value can be found by taking 10 to the power of the 0.50-probability-risk-value (since the log scale is on a base 10). From the graph, the 0.50-probability-risk-value is approximately -0.8. Thus, the median risk is calculated to be:

$$Risk_{median} = 10^{-0.8} = 0.16 \ person\text{-}rem/year$$

or 1.6 person-rem for the ten year embrittlement duration. The difference between the median and the 95th values illustrates how the uncertainty can skew the calculated values. But even though the uncertainty may result in a wide range of values, the best estimate should be used in decisionmaking due to the conservative nature of the analysis.

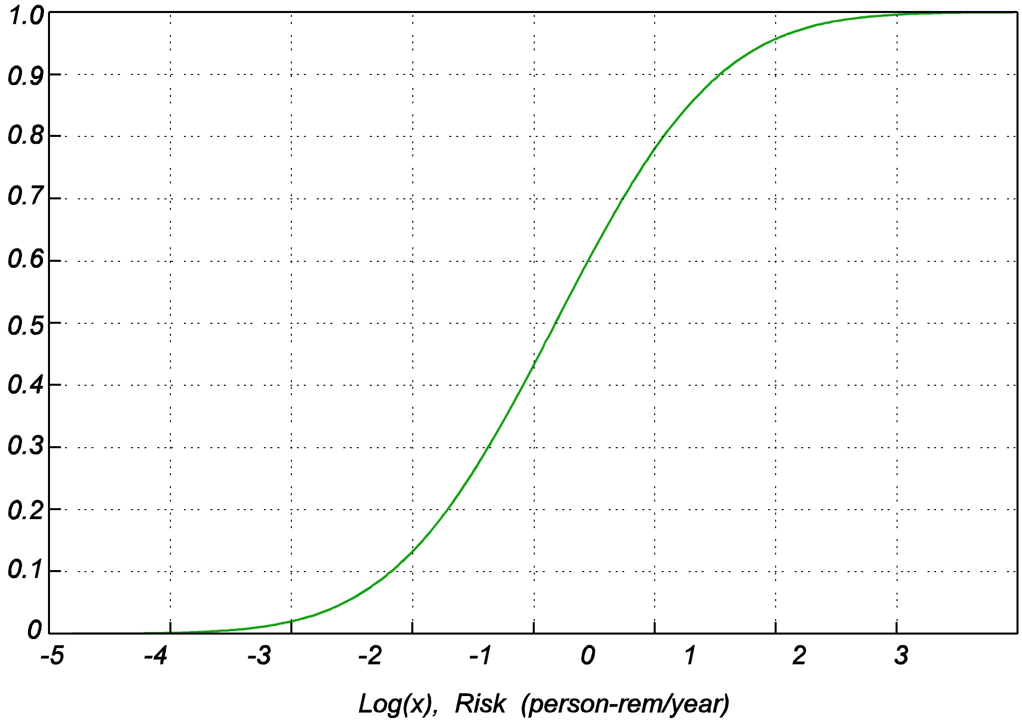

Figure A.1 : Base Case Risk Cumulative Probability Curve.

APPENDIX B - GSI-15 Risk Sensitivity Analysis

To judge how sensitive the results of the GSI-15 risk calculations (benefit evaluation) were to the values used for event tree quantification, several supplemental cases were evaluated with even more conservative estimates of failure probabilities. Seven cases were evaluated as discussed below.

Case 1 Increase the frequency of an SSE by a factor of ten. For most plants, this will have the same effect as assuming that a 0.05g earthquake will have sufficient force to potentially result in RPVS failure.

Case 2 Increase offsite dose rates by a factor of 100. This will show the potential results for a plant located in an area of high population density.

Case 3 Increase the probabilities of RPVSF and LBLOCA to 1. This will show the maximum uncertainty in the RPVS failure mechanisms.

Case 4 Increase the probabilities of RPVSF and RPSF to 1 and decrease the probability of LBLOCA to 0. This will show the maximum uncertainty in the reactor protection system failure mechanisms.

Case 5 Increase the probability of ECCSF and CSDSF to 1. This will show the maximum uncertainty involved in initiating event-induced failure of these safety systems.

Case 6 Increase the probabilities of LBLOCA and ECCSF to 1. This will show the maximum uncertainty involving the dependence of a LBLOCA and ECCS failure on RPVS failure. In other words, it simulates the pressure vessel falling sufficiently (following RPVS failure) to allow the ECCS injection lines to break or become inoperable.

Case 7 Set the probabilities of RPVSF, LBLOCA, RPSF, CSDSF, and ECCSF to 1. This allows for a worst case model of complete failure of the entire reactor protection system with the exception of the containment. This scenario should be considered to be a worst case scenario where the RPVSs and RPV supporting piping are embrittled. Following the initiating event, the subsequent shifting of the RPV results in failure of all core protection systems.

Table B.1. shows the risk results for each of the seven cases and the base case. The results for each case are given in terms of core melt frequency and expected offsite dose (person-rem) per year per plant. Also included in the table are the risks associated with ten, twenty, forty, and sixty years of cumulative operation in a condition where the RPVSs are susceptible to failure.

Table B.1. Sensitivity analysis results.

Case	Core melt frequency (per year)	Risk (per year) [person-rem]	Risk (10 years) [person-rem]	Risk (20 years) [person-rem]	Risk (40 years) [person-rem]	Risk (60 years) [person-rem]
1	$6.5x10^{-4}$	21	210	420	840	1,300
2	$8.8x10^{-5}$	290	2,900	5,800	12,000	17,000
3	$1.8x10^{-4}$	10	100	200	400	600
4	$3.5x10^{-4}$	2.7	27	54	110	160
5	$6.6x10^{-4}$	26	260	520	1,000	1,600
6	$8.8x10^{-4}$	49	490	980	2,000	2,900
7	$1.8x10^{-3}$	98	980	2,000	3,900	5,900
Base	$8.8x10^{-5}$	2.9	29	58	120	170

Four extreme cases of cost/benefit were calculated from the results of Table B.1 and the costs from Table 7. The four extreme cases were:

$$\frac{maximum\ \ cost}{minimum\ \ benefit} \qquad II.\ \ \frac{maximum\ \ c}{maximum\ \ be}$$

$$\frac{minimum\ \ cost}{minimum\ \ benefit} \qquad IV.\ \ \frac{minimum\ \ c}{maximum\ \ be}$$

For the above case, the minimum benefit was assumed to be 27 person-rem (Table B.1, case 4, for 10 years), the maximum benefit was assumed to be 17,400 person-rem (Table B.1, case 2, for 60 years), the minimum cost was assumed to be $920,000 (Table 7, Option 2, with AOSC but without replacement power), and the maximum cost was assumed to be $89M (Table 7, Option 3, with AOSC and replacement power). The results of the four extreme cost/benefit cases are presented in Table B.2.

Table B.2. Extreme Cost/Benefit results.

Case	Case description	Cost/Benefit ($/person-rem)
I	maximum cost/minimum benefit	3,300,000
II	maximum cost/maximum benefit	5,100
III	minimum cost/minimum benefit	34,000
IV	minimum cost/maximum benefit	53

APPENDIX C - Taylor Series Expansion Program Verification

To assist with the analysis contained in this report, a computer program (TSE) was used to evaluate the Taylor series expansion expressions. As a check for the program, several sample problems were entered in the program to be verified. Also, portions of the analysis in this report were hand calculated to check the numerical results. The remainder of this appendix illustrates how the Taylor series calculations are made and two sample problems are given.

Two equations from the Taylor series expansion arise depending on whether the resulting variable is calculated by a product or a summation. For the case of the product $Z = X_1 \cdot X_2 \cdot X_3 \cdot \ldots \cdot X_n$, the mean and standard deviation are found by:

$$mean\ of\ Z = \mu_Z = \mu_{X_1} \cdot \mu_{X_2} \cdot \mu_{X_3} \cdot \ldots \cdot \mu_{X_n}$$

$$standard\ deviation\ of\ Z = \sigma_Z = \left(\sum_{i=1}^{n} \left(\frac{\partial Z}{\partial X_i} \right)^2 \left(\sigma_{X_i}\right)^2 \right)^{1/2}$$

For the case of the summation, if $Z = X_1 + X_2 + X_3 + \ldots + X_n$, the mean and standard deviation are found by:

$$mean\ of\ Z = \mu_Z = \mu_{X_1} + \mu_{X_2} + \mu_{X_3} + \ldots + \mu_{X_n}$$

$$standard\ deviation\ of\ Z = \sigma_Z = \left(\sum_{i=1}^{n} (\sigma_{X_i})^2 \right)^{1/2}$$

The TSE program will calculate the mean and standard deviation for any function that can be entered into the program. The partial derivatives are numerically calculated within the program, thereby reducing the analysis time.

For the first sample problem to verify the TSE program, a problem from the statistics book Statistical Models in Engineering[a] by G. Hahn and S. Shapiro was evaluated. The problem asks to calculate the electron current for the circuit given in Figure C.1. The equation to calculate the current is:

[a] Hahn, G. J. and S. S. Shapiro, Statistical Models in Engineering, John Wiley & Sons, Inc., 1967, pp. 230-232.

$$I = V\left(\frac{1}{R_A} + \frac{1}{R_B} + \frac{1}{R_C}\right)$$

where I = current (amps)
V = voltage (volts)
R = resistance (ohms)

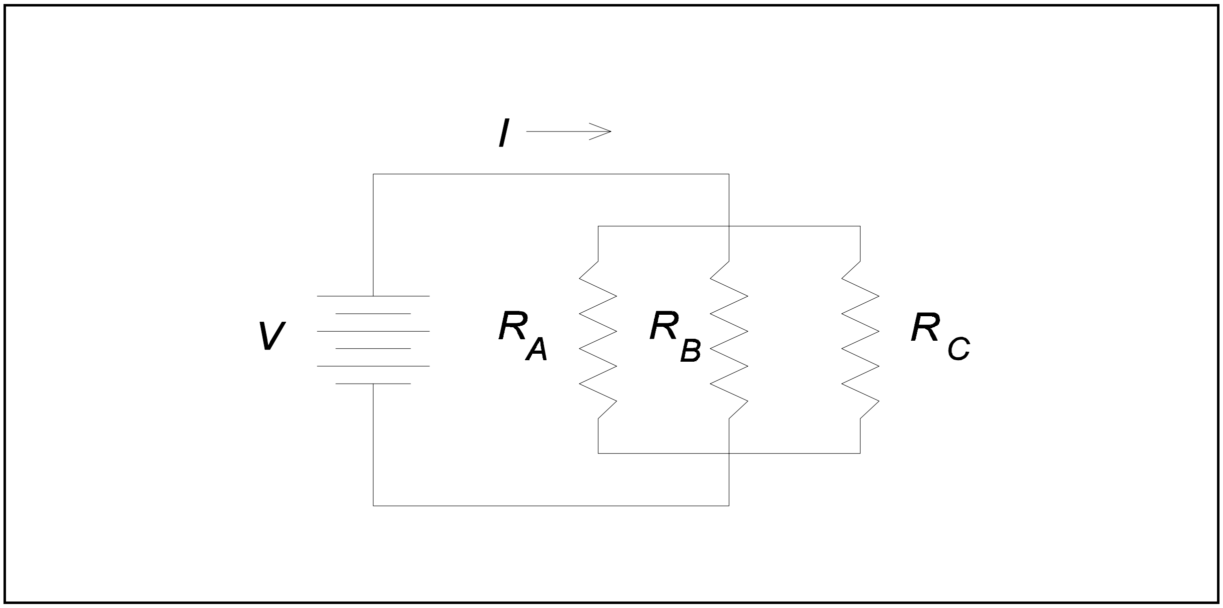

Figure C.1. Circuit Diagram for Example Problem #1.

Each of the parameters in the equation above are statistical variables. Table C.1 lists each variable with its mean and standard deviation. Hahn and Shapiro gave the answer for the current as a mean of 26.19 and a standard deviation of 1.616. The TSE program calculates the mean as 26.1 and the standard deviation as 1.61. Thus, very close agreement between the two answers is evident.

Table C.1. Variable parameters for the circuit problem.

Variable	Mean	Standard deviation
V	120	3.873
R_A	10	1
R_B	15	1
R_C	20	1.414

The second example problem is a hand calculation of the PWR 9 sequence for the analysis in this report. The PWR 9 sequence is contained within both the SSE event tree and the SBLOCA event tree (Figure 1 and Figure 2, respectively). For the SSE event tree, the PWR 9 sequence can be written as:

$$Z_1 = SSE \cdot RPVSF \cdot LBLOCA \cdot \overline{ECCSF} \cdot \overline{RCF}$$

where the bar over the event denotes the compliment of the event. Before evaluating this sequence, the event parameters must be known. From Appendix A, the parameters are shown in Table C.2. It should be pointed out that the numerically calculated results are shown in this Appendix with three significant digits for calculational purposes only.

Table C.2. Variable parameters for PWR 9 sequence.

Event	Mean	Standard deviation
SSE	$1.25x10^{-3}$	$3.5x10^{-2}$
SBLOCA	$5.0x10^{-4}$	$2.2x10^{-2}$
RPVSF	$5.0x10^{-1}$	$2.0x10^{-1}$
LBLOCA	$5.0x10^{-1}$	$2.0x10^{-1}$
ECCSF	$1.0x10^{-1}$	$1.0x10^{-1}$
RCF	$1.0x10^{-3}$	$5.0x10^{-3}$

From page C-1, the mean and standard deviation for the equation Z_1 can be calculated as:

$$\mu_{Z_1} = \mu_{SSE} \cdot \mu_{RPVSF} \cdot \mu_{LBLOCA} \cdot \mu_{\overline{ECCSF}} \cdot \mu_{\overline{RCF}}$$

$$\sigma_{Z_1} = \left[\sum_{i=1}^{N} \left(\frac{\partial Z_1}{\partial X_i} \right)^2 (\sigma_{X_i})^2 \right]^{1/2}$$

Evaluating the mean results in:

$$\mu_{Z_1} = (1.25x10^{-3})\,(0.5)\,(0.5)\,(1 - 0.1)\,(1 - (1.0x10^{-3}))$$

$$= 2.81x10^{-4}$$

Taking the equation for the standard deviation, each term will be written out and evaluated separately. Thus, we find:

$$\sigma_{Z_1}^{\ 2} = \left(\frac{\partial Z_1}{\partial SSE}\right)^2 (\sigma_{SSE})^2 + \left(\frac{\partial Z_1}{\partial RPVSF}\right)^2 (\sigma_{RPVSF})^2 + \left(\frac{\partial Z_1}{\partial LBLOCA}\right)^2 (\sigma_{LBLOCA})^2$$

$$+ \left(\frac{\partial Z_1}{\partial \overline{ECCSF}}\right)^2 (\sigma_{\overline{ECCSF}})^2 + \left(\frac{\partial Z_1}{\partial \overline{RCF}}\right)^2 (\sigma_{\overline{RCF}})^2$$

Evaluating the first term in the equation above yields:

$$\left(\frac{\partial Z_1}{\partial SSE}\right)^2 (\sigma_{SSE})^2 = (\mu_{RPVSF} \cdot \mu_{LBLOCA} \cdot \mu_{\overline{ECCSF}} \cdot \mu_{\overline{RCF}})^2 (\sigma_{SSE})^2$$

Substituting the appropriate mean values results in:

$$\left(\frac{\partial Z_1}{\partial SSE}\right)^2 (\sigma_{SSE})^2 = \left[(0.5)\,(0.5)\,(1-0.1)\,(1-(1.0x10^{-3}))\right]^2 (3.5x10^{-2})^2$$

$$= 6.19x10^{-5}$$

The four remaining terms are:

$$\left(\frac{\partial Z_1}{\partial RPVSF}\right)^2 (\sigma_{RPVSF})^2 = (\mu_{SSE} \cdot \mu_{LBLOCA} \cdot \mu_{\overline{ECCSF}} \cdot \mu_{\overline{RCF}})^2 (\sigma_{RPVSF})^2$$

$$= \left[(1.25x10^{-3})\,(0.5)\,(1-0.1)\,(1-(1.0x10^{-3}))\right]^2 (0.2)^2$$

$$= 1.26x10^{-8}$$

$$\left(\frac{\partial Z_1}{\partial LBLOCA}\right)^2 (\sigma_{LBLOCA})^2 = (\mu_{SSE} \cdot \mu_{RPVSF} \cdot \mu_{\overline{ECCSF}} \cdot \mu_{\overline{RCF}})^2 (\sigma_{LBLOCA})^2$$

$$= \left[(1.25x10^{-3})\,(0.5)\,(1-0.1)\,(1-(1.0x10^{-3}))\right]^2 (0.2)^2$$

$$= 1.26x10^{-8}$$

$$\left(\frac{\partial Z_1}{\partial \overline{\overline{ECCSF}}}\right)^2 (\sigma_{\overline{ECCSF}})^2 = \left(-(\mu_{SSE} \cdot \mu_{RPVSF} \cdot \mu_{LBLOCA} \cdot \mu_{\overline{RCF}})\right)^2 (\sigma_{ECCSF})^2$$

$$= \left[-(1.25x10^{-3})\,(0.5)\,(0.5)\,(1-(1.0x10^{-3}))\right]^2 (0.1)^2$$

$$= 9.75x10^{-10}$$

$$\left(\frac{\partial Z_1}{\partial \overline{RCF}}\right)^2 (\sigma_{\overline{RCF}})^2 = \left(-(\mu_{SSE} \cdot \mu_{RPVSF} \cdot \mu_{LBLOCA} \cdot \mu_{\overline{ECCSF}})\right)^2 (\sigma_{RCF})^2$$

$$= \left[-(1.25x10^{-3})\,(0.5)\,(0.5)\,(1-0.1)\right]^2 (5.0x10^{-3})^2$$

$$= 1.98x10^{-12}$$

From the five above terms, the standard deviation of Z_1 is found by:

$$\sigma_{Z_1} = \left(6.19x10^{-5} + 1.26x10^{-8} + 1.26x10^{-8} + 9.75x10^{-10} + 1.98x10^{-12}\right)^{1/2}$$

$$= 7.87x10^{-3}$$

Now, the PWR 9 sequence from the SBLOCA event tree will be analyzed in a similar manner. The PWR 9 sequence for the SBLOCA event tree can be written as:

$$Z_2 = SBLOCA \cdot RPVSF \cdot LBLOCA \cdot \overline{ECCSF} \cdot \overline{RCF}$$

The mean and standard deviation of the SBLOCA PWR 9 sequence are:

$$\mu_{Z_2} = \mu_{SBLOCA} \cdot \mu_{RPVSF} \cdot \mu_{LBLOCA} \cdot \mu_{\overline{ECCSF}} \cdot \mu_{\overline{RCF}}$$

$$\sigma_{Z_2}^{\;2} = \left(\frac{\partial Z_2}{\partial SBLOCA}\right)^2 (\sigma_{SBLOCA})^2 + \left(\frac{\partial Z_2}{\partial RPVSF}\right)^2 (\sigma_{RPVSF})^2 + \left(\frac{\partial Z_2}{\partial LBLOCA}\right)^2 (\sigma_{LBLOCA})^2$$

$$+ \left(\frac{\partial Z_2}{\partial \overline{ECCSF}}\right)^2 (\sigma_{\overline{ECCSF}})^2 + \left(\frac{\partial Z_2}{\partial \overline{RCF}}\right)^2 (\sigma_{\overline{RCF}})^2$$

Calculating the mean value results in:

$$\mu_{Z_2} = (5.0x10^{-4})\,(0.5)\,(0.5)\,(1 - 0.1)\,(1 - (1.0x10^{-3}))$$

$$= 1.12x10^{-4}$$

Calculating the five terms for use in the standard deviation equation results in:

$$\left(\frac{\partial Z_2}{\partial SBLOCA}\right)^2 (\sigma_{SBLOCA})^2 = 2.45x10^{-5}$$

$$\left(\frac{\partial Z_2}{\partial RPVSF}\right)^2 (\sigma_{RPVSF})^2 = 2.02x10^{-9}$$

$$\left(\frac{\partial Z_2}{\partial LBLOCA}\right)^2 (\sigma_{LBLOCA})^2 = 2.02x10^{-9}$$

$$\left(\frac{\partial Z_2}{\partial \overline{ECCSF}}\right)^2 (\sigma_{\overline{ECCSF}})^2 = 1.56x10^{-10}$$

$$\left(\frac{\partial Z_2}{\partial \overline{RCF}}\right)^2 (\sigma_{\overline{RCF}})^2 = 3.16x10^{-13}$$

Therefore, the standard deviation of Z_2 is found by:

$$\sigma_{Z_2} = \left(2.45x10^{-5} + 2.02x10^{-9} + 2.02x10^{-9} + 1.56x10^{-10} + 3.16x10^{-13}\right)^{1/2}$$

$$= 4.95x10^{-3}$$

Now, the total PWR 9 sequence probability is calculated by adding the SSE results to the SBLOCA results:

$$PWR\ 9_{TOTAL} = PWR\ 9_{SSE} + PWR\ 9_{SBLOCA}$$

From page C-1, when two variables are added, the mean and standard deviation can be calculated from:

$$\mu_{PWR\ 9_{TOTAL}} = \mu_{PWR\ 9_{SSE}} + \mu_{PWR\ 9_{SBLOCA}}$$

$$= 2.81x10^{-4} + 1.12x10^{-4} = 3.93x10^{-4}$$

$$\sigma_{PWR\ 9_{TOTAL}} = \left((\sigma_{PWR\ 9_{SSE}})^2 + (\sigma_{PWR\ 9_{SBLOCA}})^2\right)^{1/2}$$

$$= \left((7.87x10^{-3})^2 + (4.95x10^{-3})^2\right)^{1/2}$$

$$= 9.30x10^{-3}$$

Table A.2 lists the calculated sequence end states from the TSE program. For the total PWR 9 sequence, the calculated mean is $3.9x10^{-4}$ and the calculated standard deviation is $9.3x10^{-3}$. These calculated results confirm the above hand calculated values.

APPENDIX D - GSI-15 Cost/Benefit Ratio Graphs

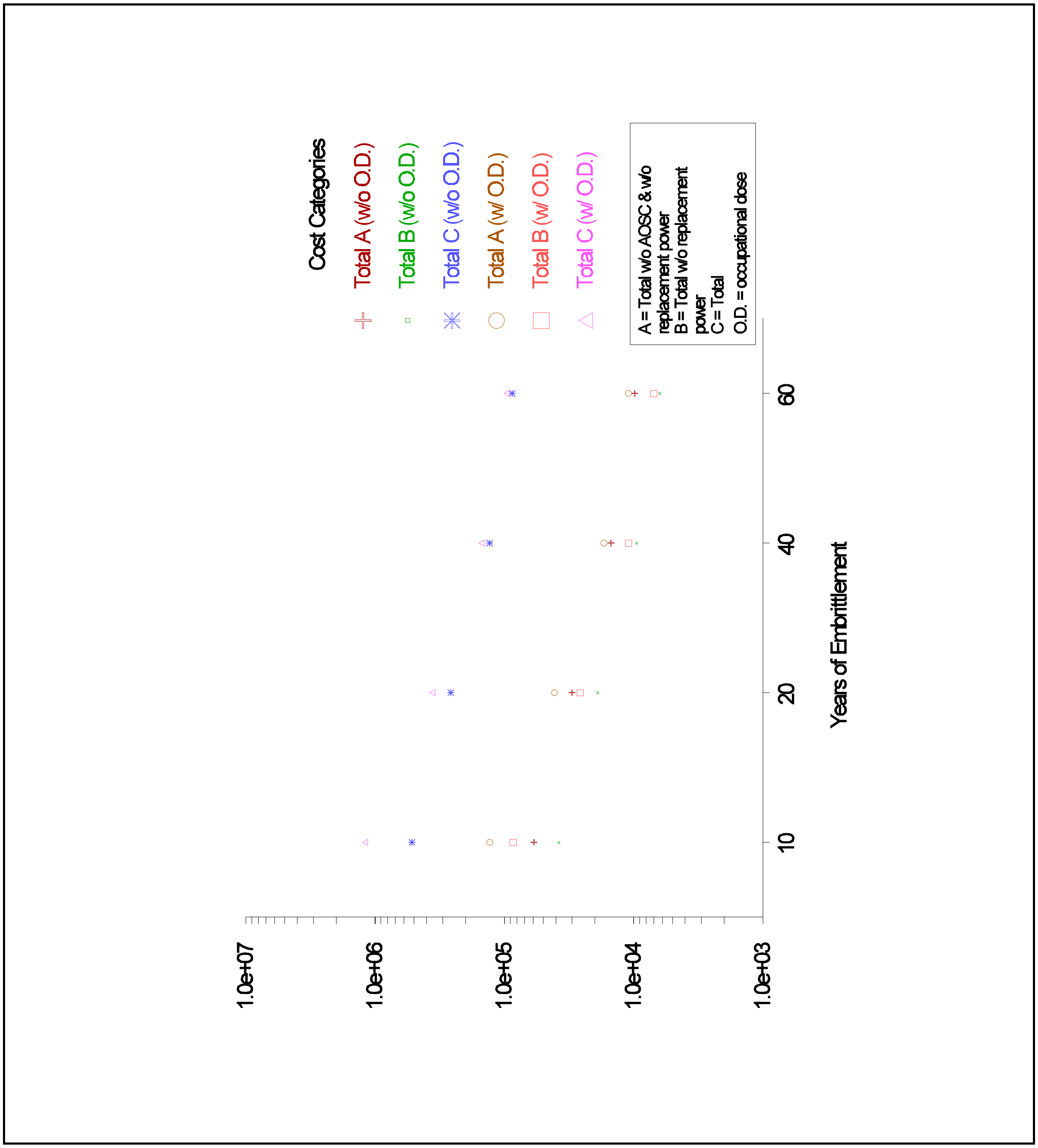

Figure D.1. Option 1 Cost/Benefit Ratios.

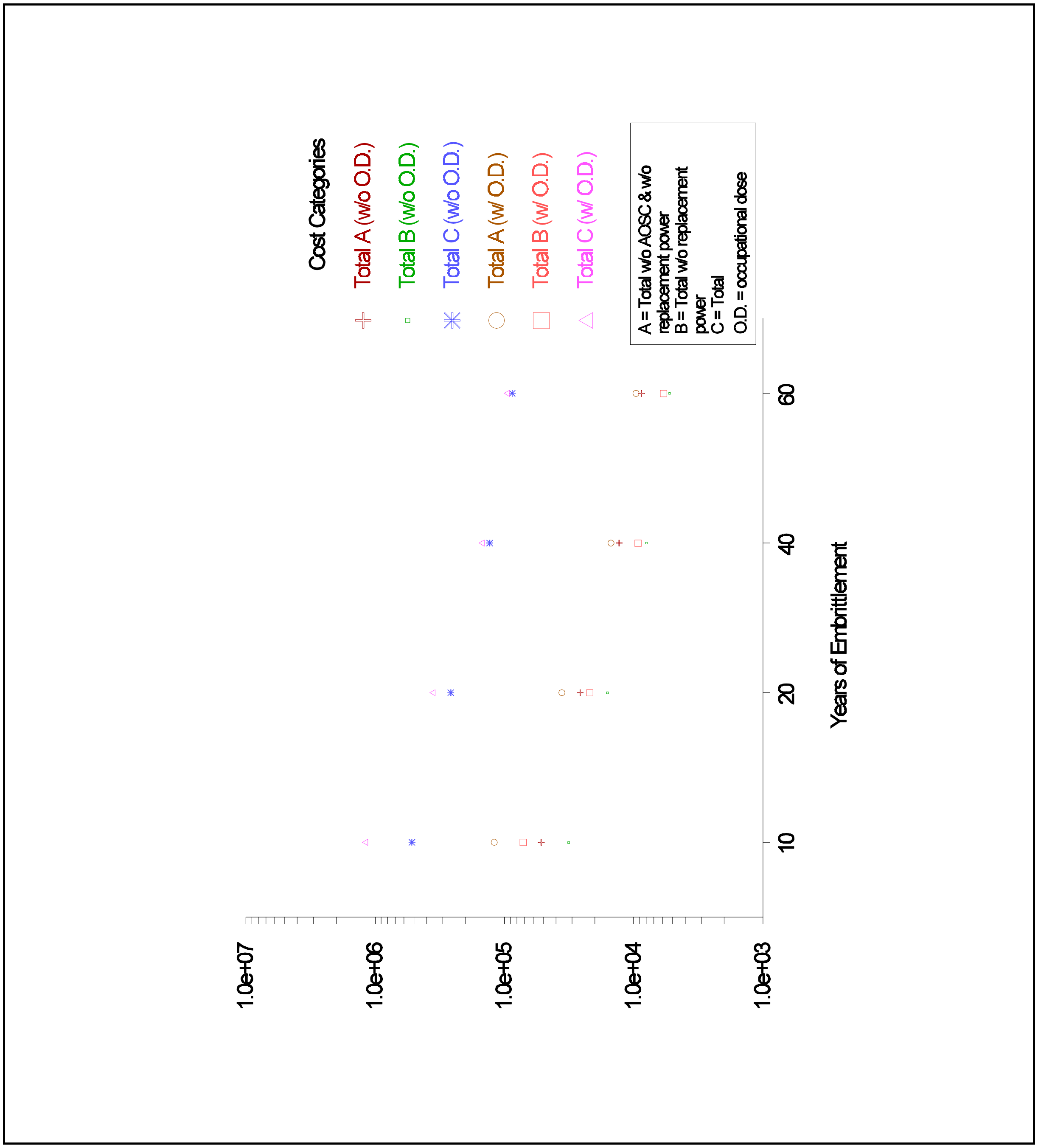

Figure D.2. Option 2 Cost/Benefit Ratios.

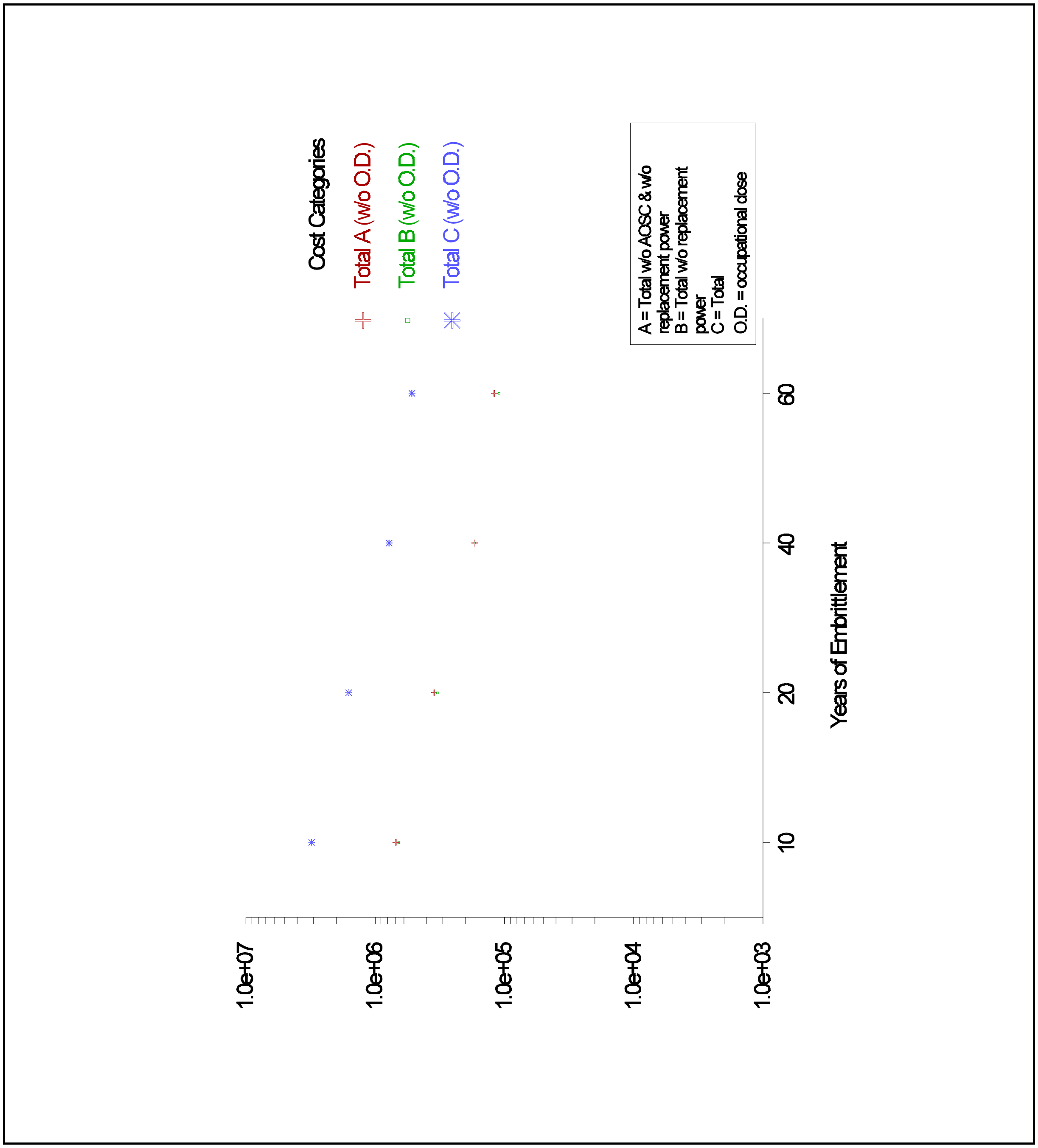

Figure D.3. Option 3 Cost/Benefit Ratios.

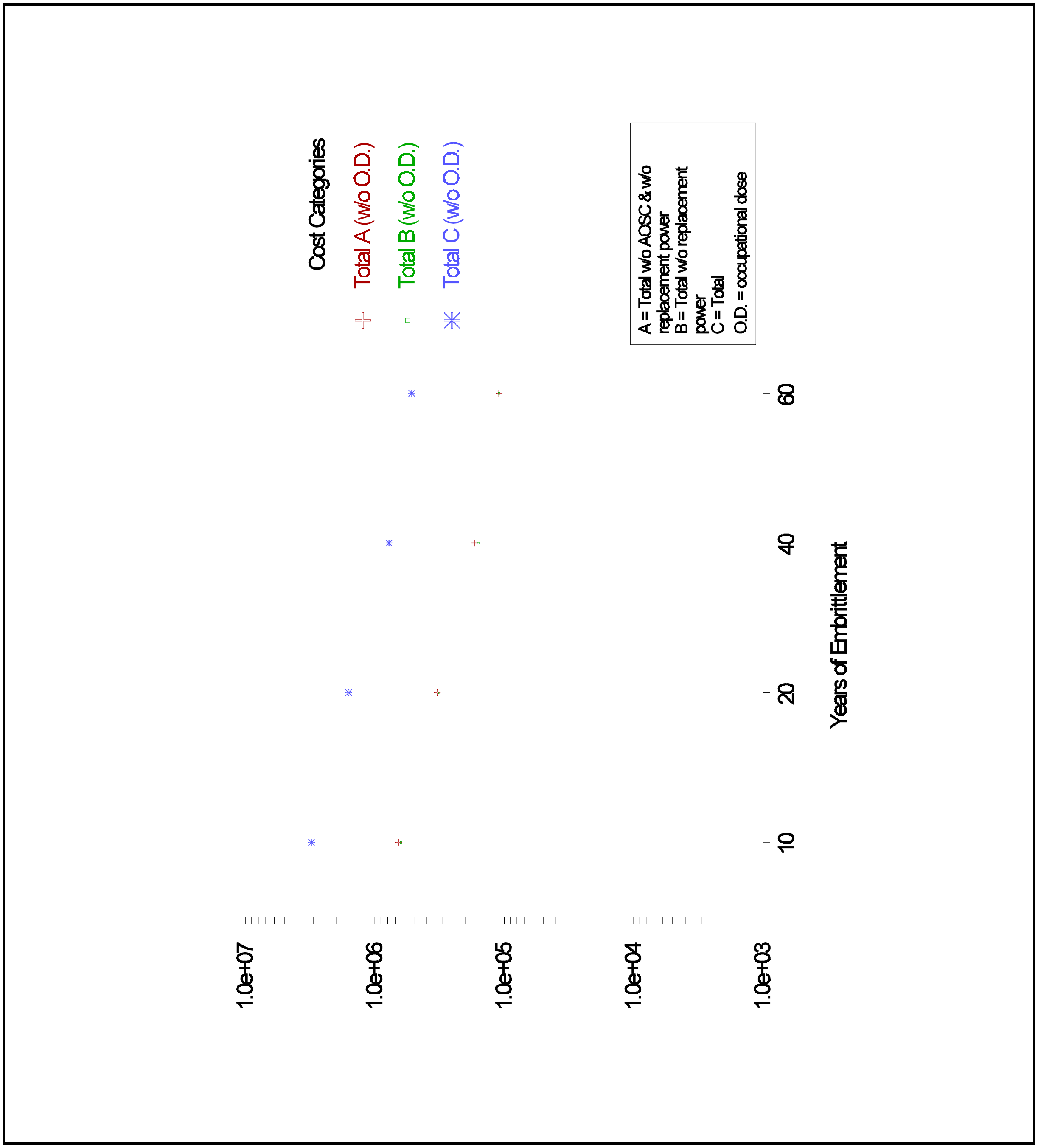

Figure D.4. Option 4A Cost/Benefit Ratios.

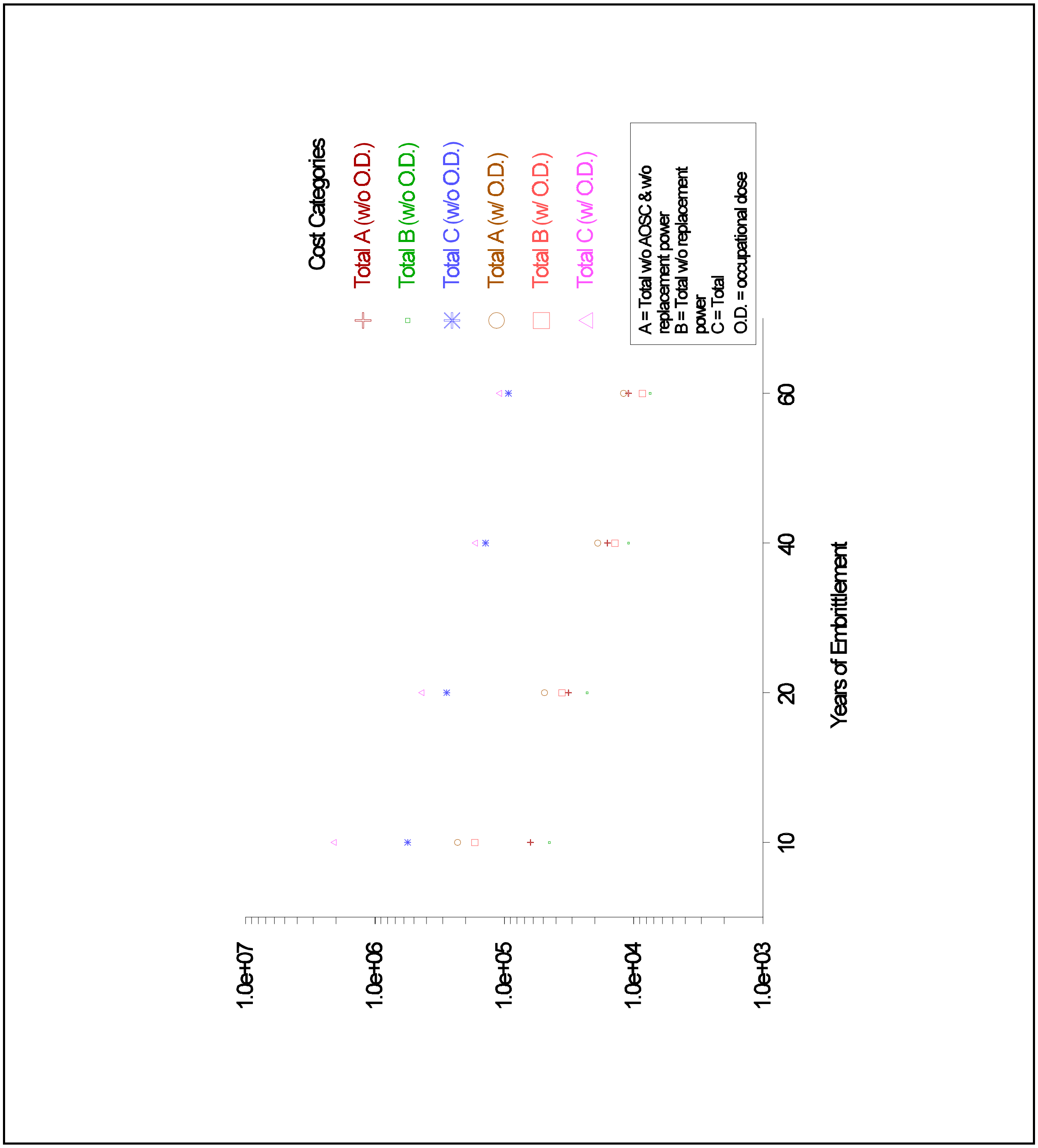

Figure D.5. Option 4B Cost/Benefit Ratios.

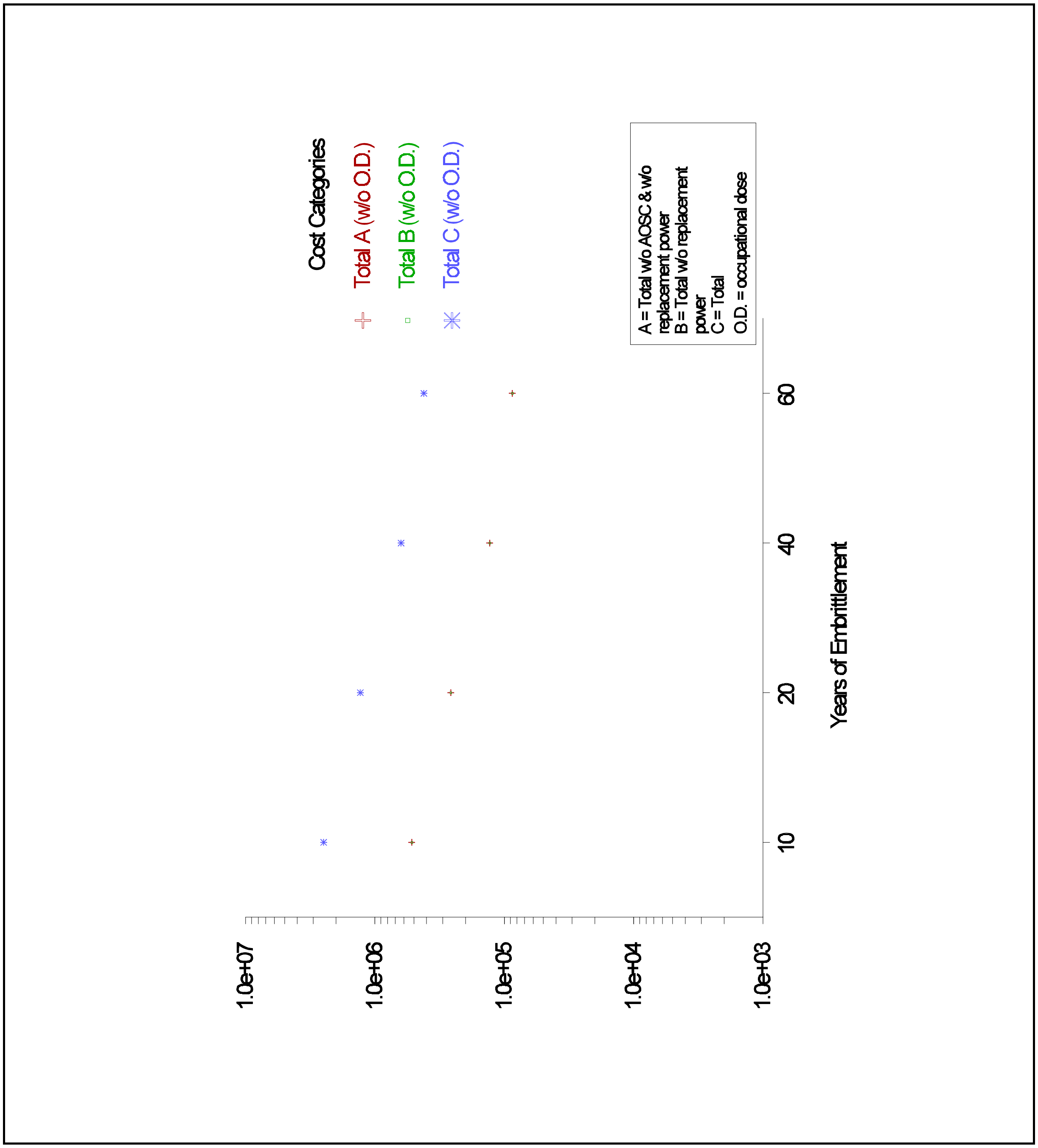

Figure D.6. Option 5 Cost/Benefit Ratios.